Response to the U.S. Nuclear Regulatory Commission Request for Additional Information on the Draft Waste Incidental to Reprocessing Evaluation for the West Valley Demonstration Project Vitrification Melter

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Acronyms and Abbreviations

ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
DOE	Department of Energy
HLW	high-level waste
LLW	low-level waste
MEP	maximum extent practical
NRC	Nuclear Regulatory Commission
PA	performance assessment
RAI	request for additional information
VAST	Verification Analytical Sample Tracking
WCS	Waste Control Specialists
WIR	waste incidental to reprocessing
WVDP	West Valley Demonstration Project
WVES	West Valley Environmental Services
WVNSCO	West Valley Nuclear Services Company

Units

Bq	Becquerel
Ci	curie
cm	centimeter
cm^3	centimeter cubed
g	gram [mass]
hr	hour
kg	kilogram
L	liter
m	meter
m^3	cubic meter
µCi	0.000001 curie
R	Roentgen

INTRODUCTION

The U.S. Department of Energy (DOE) is consulting with the U.S. Nuclear Regulatory Commission (NRC) on the draft waste-incidental-to-reprocessing evaluation for the West Valley Demonstration Project (WVDP) vitrification melter (DOE-WV 2011), which was prepared in accordance with DOE Manual 435.1-1, *Radioactive Waste Management*. To this end DOE submitted the draft evaluation to the NRC for review on March 8 2011.

NRC provided the formal request for additional information (RAI, NRC 2011a) on May 10, 2011 with 17 comments.

DOE is providing responses to the RAI comments and making some related changes to the waste-incidental-to-reprocessing evaluation prior to finalizing it to ensure that management and disposal of the vitrification melter is protective of human health and the environment. The DOE responses are provided in the following format:

Number: The NRC number is specified.

Subject: The subject as identified by NRC is listed.

Basis: The basis for the comment as described by NRC is reproduced in its entirety.

Path forward: The recommended path forward provided by NRC with each comment is reproduced in its entirety. In some cases multiple requests for information have been numbered for clarity.

DOE response: The DOE response addresses the comment, providing explanatory information on the basis for the response.

Conclusions and changes to the draft evaluation: Conclusions from consideration of the NRC comment and the related response and any changes to be made to the draft evaluation are identified.

References are cited where appropriate, with a reference list provided at the end of the responses. Note that some NRC reference citations were changed for consistency with the reference list, which includes a number of other documents.

Four attachments are provided to supplement information in the responses.

Number: IN-1

Subject: The draft WIR evaluation does not provide sufficient technical information to support homogeneity of the waste form.

Basis: NUREG-1854, Section 3.1, advises that the NRC determine whether the waste samples adequately represent heterogeneity of the waste (NRC 2007). The draft WIR evaluation states on page 24, "The characterization for residual activity inside the Vitrification Melter was based on measured gamma dose rates and analytical data from two samples of residual glass" (DOE-WV 2011). DOE provides the following basis to support homogeneity of the waste on page 39 of the draft WIR evaluation: (1) that the waste form was made homogeneous through the mixing in the Melter Feed Hold Tank and, (2) that the molten state of the glass achieved through the high process heats contributed to homogeneity. These reasons do not sufficiently address the potential for hardened glass in the Melter.

DOE suggests the potential for hardened glass in the Melter over time that plugged small spaces such as joints around nozzles, pipes used for airlifting, and in the discharge port (Brooks, 1993). Depending on the variability in the waste streams being fed into the Melter over time, there is potential for pockets of hardened glass to contribute variability to the glass content remaining in the Melter. For example, residual, hardened glass deposited in Melter might vary over space and time (e.g., glass that hardens in refractory material joints over time). Also, residual contamination on the roof of the Melter that accumulated due to vaporization of radioactivity is expected to have a different radionuclide mix compared to the contamination in the Melter. Contamination deposited in piping may also have accumulated over time. If this variability exists within the Melter, the glass samples taken from the evacuated canisters may not be representative of the glass remaining in the Melter because the glass remaining may represent a stagnant portion of glass that is more difficult to retrieve through normal operations

Path Forward DOE should provide basis for why glass sample analytical data scaling factors are representative of all of the contamination remaining within the Melter. Provide any additional technical information to support homogeneity of the glass (e.g., samples with low variability, gamma readings that are fairly consistent after taking into account differences in geometry). Data that could potentially support the homogeneity assumption include periodic samples from the concentrator feed make-up tank or Melter Feed Hold Tank, or additional description of the chemical processes taking place and their effects on homogeneity – (e.g., studies on the waste form itself).

> **DOE response:** Further evaluation of scaling factors in connection with this comment resulted in confirming that the glass sample scaling factors are conservative or representative of the radiological characteristics of all of the contamination remaining in the vitrification melter, including the residual glass in the plugged glass exit chamber. The basis for this conclusion is explained below.
>
> This response begins with a summary of information separately provided to NRC and some background information to help place the technical basis for the conclusion into context.
>
> **Additional Information Provided to NRC**
>
> Information related to the homogeneity of the material in the vitrification melter in the form of analytical data from samples taken from the concentrator feed makeup tank and gamma dose rates on the outside of the high-level waste (HLW) canisters has been provided separately to NRC. Attachment 1 provides the graphical representation of the variations in the canister dose rates that was provided.

Background Information

It is useful for perspective to consider background information on the following matters related to the vitrification program before the basis for the conclusion about the scaling factors is discussed:

- System testing,
- Vitrification operations,
- Plugging of the vitrification melter primary glass exit chamber,
- Completing vitrification,
- Radionuclide concentration and dose rate data consistency, and
- Feed tank and vitrification melter processes that promoted homogeneity.

System Testing. Full-scale testing of the vitrification system design was accomplished from 1984 through 1989 using an actual vitrification melter and the same concentrator feed makeup tank and melter feed hold tank to be used in the HLW vitrification. A nonradioactive simulated slurry material was used in these tests and some 150,000 kilograms of glass were produced. The vitrification melter used in this testing was disassembled and inspected (Brooks 1993).

Prior to the start of HLW vitrification, extensive component and system testing was performed with a second vitrification melter that is the subject of the draft evaluation and the other equipment to be used for HLW vitrification. This testing included integrated system runs with nonradioactive slurry that produced canisters of glass. Nonradioactive slurry was mixed in the concentrator feed makeup tank, sent to the melter feed hold tank, and processed through the vitrification melter[1]. It is assumed that most of glass that accumulated in fissures in the refractory of the new vitrification melter came from this process based on conditions found in the test melter (Brooks 1993).

Vitrification Operations. HLW solidification involved transferring the HLW slurry mixture from Tank 8D-2 to the concentrator feed makeup tank, where it was blended with batches of cold chemicals (including glass formers), concentrated, transferred to the melter feed hold tank, continuously fed to the vitrification melter, and poured into stainless steel canisters to produce the final HLW glass waste form. During the period from July 1996 through August 2002, a total of 275 HLW canisters were produced.

The basic process used to produce canisters of HLW glass began by combining the radioactive species captured in the zeolite media during HLW pretreatment with the HLW sludge, blending in recycled waste liquids resulting from off-gas treatment, and transferring the HLW slurry from Tank 8D-2 to the concentrator feed makeup tank, where the first step of HLW processing took place. This processing step involved taking slurry samples and sending them to the onsite Analytical and Process Chemistry Laboratory for chemical and radiochemical analysis to determine the exact glass-making recipe needed to process the slurry into qualified HLW glass that meets the waste acceptance specification (DOE 1996).

While slurry samples were analyzed, the batch of slurry in the concentrator feed makeup tank was concentrated to remove excess water. After the batch recipe was determined and a pre-mix of chemicals prepared, the chemicals were transferred into the concentrator feed makeup tank, mixed with the concentrated slurry and this material sampled again to ensure that the

[1] The first radioactive batch was designated batch 10 for reference purposes. There were no batches 6 through 9.

glass will meet the waste acceptance specification (DOE 1966). (The final sample results were used in the evaluation.)

The next step in HLW processing involved transferring feed to the vitrification melter, allowing water to evaporate, salts to decompose, and remaining solids to calcine. Inside the vitrification melter, calcined wastes and glass-formers melted and fused into a glass pool where they homogenized. During vitrification melter operation, homogenized molten glass was periodically airlifted into a canister held in position under the vitrification melter glass exit chamber. Figure 1 shows the conceptual process in a simplified fashion.

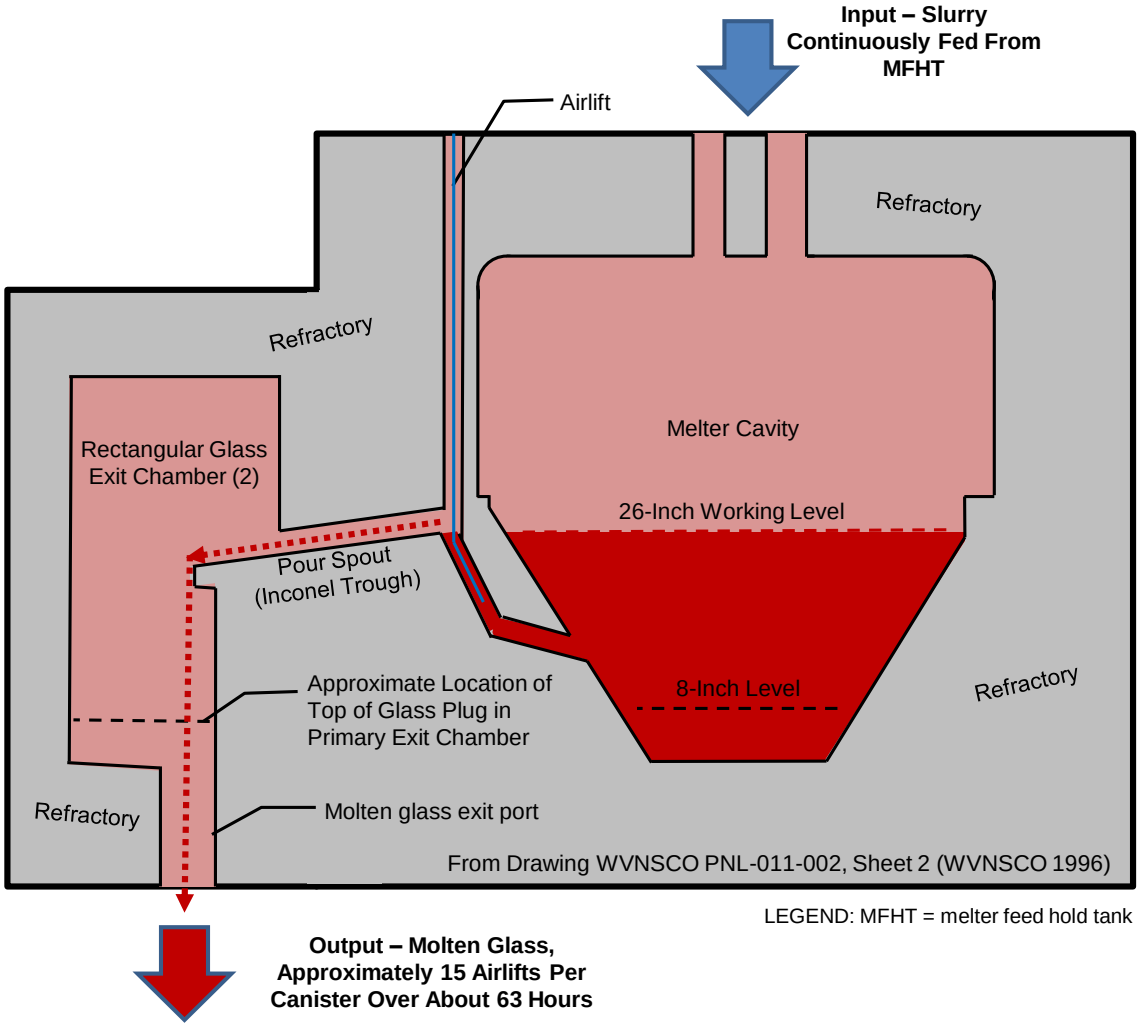

Figure 1. Melter General Operational Concept

Maintenance was required from time to time during operations and this maintenance was performed without shutting down the vitrification melter. For example, the agitators in both the concentrator feed makeup tank and the melter feed hold tank were replaced due to excessive wear. And late in the vitrification program, a failure occurred in the primary discharge chamber when the glass exit chamber became plugged with glass.

Plugging of the Primary Glass Exit Port. The glass exit port and the lower part of the glass exit chamber became plugged on January 4, 2002. At this time, concentrator feed makeup tank batch 75, which included the last of the material from Tank 8D-2, was being fed into the

vitrification melter. (Note that the vitrification melter molten glass pool typically contained material from more than one batch of slurry feed material, as discussed later.) Molten glass was being airlifted into canister number 266, the fourth canister being filled while batch 75 was being fed into the vitrification melter. When this canister was approximately one-third full, the glass pour abruptly ceased. This was not an unexpected failure and the other glass exit path was heated up, brought on line in early May 2002, and used until completion of the vitrification program.

Completing Vitrification. Two additional batches of material were sent to the vitrification melter after batch 75. Batch 76 and 77 were made up of vitrification system flush solutions consisting of dilute nitric acid, water, and residual radioactive waste removed from the vitrification system equipment by the decontamination process, along with glass formers to facilitate vitrification.

Data Consistency. The sample analytical data from the concentrator feed makeup tank demonstrate a reasonable degree of homogeneity of the glass over time. The first HLW transfer from Tank 8D-2 to the Vitrification Facility took place on June 24, 1996. The spreadsheet for the concentrator feed makeup tank data shows this to be batch 10; the previous feed material batches were nonradioactive simulated slurry used in system testing. The last feed material batch sent to the vitrification melter was batch 77.

The spreadsheet provided to NRC includes calculated ratios of Cs-137 to Sr-90, which provide a measure of the homogeneity of the feed material over the course of the vitrification program in terms of radionuclide distributions. These data show ratios in the 0.6 to 1.6 range with one outlier until batch 67. Later batches show higher ratios.

The gamma dose rates measured on the filled canisters averaged about 2620 R/hr, with variations as shown in Attachment 1. These dose rates were generally very consistent over time. Only 13 canisters were outside two sigma of the average and these 13 canisters were all produced between December 1999 and May 2001. The remaining canisters all had dose rates within a factor of 1.8 to the average. Dose rates measured in January 2003 on the exterior of the vitrification melter at the nozzles were also reasonably consistent, with Nozzle A reading 748 R/hr, Nozzle E reading 660 R/hr and Nozzle BB Reading 700 R/hr. (WMG 2004b)

Note that there is not a direct correlation between the concentrations of Cs-137, the dominant gamma-emitting radionuclide measured in samples taken from the concentrator feed makeup tank, and the measured canister dose rate as discussed in the response to comment MEP-4 below.

No waste form study that would be helpful in evaluating consistency in the melter feed material is available. However, overall, the data show reasonable consistency in the feed material over time except for the lower radionuclide concentrations in the last two batches containing decontamination solutions.

Feed Tank and Vitrification Melter Processes that Promoted Homogeneity. The material fed to the vitrification melter was consistently well mixed. Both the concentrator feed makeup tank and the melter feed hold tank contained mixing paddles that were used to ensure that the slurry and its additives were thoroughly mixed before being fed to the vitrification melter. The material in the melter feed hold tank was agitated on an essentially continuous basis during all processing operations (DesCamp and McMahon 1996).

The vitrification melter operated continuously from the beginning to the end of the vitrification program. Natural convection within the vitrification melter molten glass pool from heat input ensured a homogeneous mixture of the calcined waste and glass formers. Filling the 275 canisters with glass was a deliberate process that typically involved 10 to 15 airlifts of molten glass taking approximately 63 hours to fill each canister. (DesCamp and McMahon 1996).

The vitrification melter glass pool level was maintained near the 26 inch level throughout the vitrification period.[2] The vitrification melter capacity (227 gallons) was approximately the same as the volume of glass poured into a single canister, which was typically filled to about 90 percent of its capacity. Given the volume similarities, the slow rate of feed material entering the vitrification melter, and the multiple airlifts required to fill a single canister, the composition of the glass pool changed slowly over time as additional batches of HLW slurry were added to the vitrification melter.

Basis for Use of Glass Sample Scaling Factors for Glass in the Vitrification Melter Cavity

As explained in the draft evaluation, it was assumed for estimating purposes that three parts of the vitrification melter contain residual glass:

- An estimated 300 kilograms in the vitrification melter cavity,
- An estimated 99 kilograms in the plugged glass exit port, and
- An estimated 26 kilograms on melter cavity surfaces and in refractory material fissures.

The dose-to-curie method used to estimate the radioactivity in this glass (WMG 2004b) involved calculating the amount of Cs-137 in each of the three areas using measured dose rates and estimating the amounts of the other key radionuclides by using activity scaling factors. The scaling factors were based on analytical data from glass shard samples collected from the two evacuated canisters.

The basis for use of these scaling factors for the hardened glass remaining in the vitrification melter cavity is simply that the glass shard samples were taken from the same material. The snorkel used to suck the molten glass from the vitrification melter into the evacuated canisters reached near the bottom of the cavity, so it is clear that the samples were taken from the same material and are representative of the glass at the bottom of the vitrification melter cavity[3].

Basis for Use of Glass Sample Scaling Factors for Glass in Refractory Fissures

The Cs-137 activity in glass that seeped into fissures in the refractory material was assumed to contain approximately 12 curies (WMG 2004b). This material likely came from the early vitrification melter runs with batches of nonradioactive simulated slurry since this material would have entered the fissures. Therefore, using the glass sample scaling factors to estimate the radioactivity in this material is conservative.

Basis For Use of Glass Sample Scaling Factors for Glass in the Plugged Exit Port

[2] The operating procedure (WVNSCO 2002b) specified that airlift operations would be performed when the glass level approached 0.7 inch above the glass level at the end of the last airlift or if dripping occurred. The level was measured by a level detection instrument and levels were recorded for every canister produced and at the start of each airlift.

[3] The snorkels were positioned differently for the two evacuated canisters. Drawings (WVNSCO 2002d and WVNSCO 2002e) and the completed work package (WVNSCO 2002f) show that the lower end of the snorkel was positioned so it would draw molten glass from a point approximately six inches from the bottom of the cavity for first evacuated canister and two inches from the bottom of the cavity for the second evacuated canister.

To evaluate whether use of the glass sample scaling factors was appropriate for the hardened glass in the plugged glass exit chamber, the glass sample scaling factors have been compared to scaling factors developed using analytical data from the concentrator feed makeup tank samples from batch 74 and batch 75, the sources of most of the material in the vitrification melter when the glass exit port became plugged[4]. Table 1 shows this comparison for selected radionuclides.[5]

Table 1. Radionuclide Scaling Factors (Ratios to Cs-137)

Radionuclide	Scaling Factor Basis		
	Glass Sample Data(1)	Batch 74 Data(2)	Batch 75 Data(3)
Sr-90	5.73E-02	4.00E-02	7.47E-02
Np-237	1.41E-06	3.25E-07	6.13E-07
Pu-238	1.57E-04	5.48E-05	1.09E-04
Am-241	6.84E-04	2.19E-04	3.28E-04

NOTES: (1) From WMG 2004b, Exhibit 1.
(2) From concentrator feed makeup tank analytical data shown in the spreadsheet provided to NRC (WVES 2011).
(3) Based on Verification Analytical System Tracking (VAST) sample 01-2498 (WVNSCO 2002a).

Table 1 shows that the glass sample scaling factors are higher (more conservative) than the batch 74 and batch 75 scaling factors with one exception: the Sr-90 scaling factor for batch 75. The Sr-90 scaling factor from the glass sample data is essentially the same as the average of the other two Sr-90 scaling factors (5.73E-02 vs. 5.735E-02). Based on this comparison, the use of the glass sample scaling factors for the residual material in the plugged glass exit port is appropriate.

Alternate Estimating Method

Consideration of the comparability of the different scaling factors led to consideration as to whether the dose-to-curie estimating method itself (WMG 2004b) was inherently conservative. One measure of its conservatism would be a comparison with the results of a different estimating method involving multiplying the radionuclide concentrations in the glass samples by the estimated mass of the residual glass in the vitrification melter (425 kg). Table 2 shows this comparison.

Table 2. Vitrification Melter Total Activity Estimates (in Curies as of October 1, 2004)

Nuclide	Dose-to-Curie(1)	Concentration x Mass(2)	Nuclide	Dose-to-Curie(1)	Concentration x Mass(2)
C-14	2.12E-02	1.04E-02	U-238	2.25E-03	1.10E-03
K-40	8.19E-02	4.01E-02	Np-237	6.20E-03	3.03E-03
Mn-54	8.57E-02	4.13E-02	Pu-238	6.84E-01	3.35E-01
Co-60	8.33E-02	4.10E-02	Pu-239	1.59E-01	7.77E-02
Sr-90	2.47E+02	1.21E+02	Pu-240	NA	5.93E-02
Zr-95	1.65E+00	8.64E-01	Pu-241	3.12E+00	1.53E+00
Tc-99	1.11E-02	5.42E-03	Pu-242	NA	1.12E-05

[4] Analysis of input and output for the vitrification melter suggests that scaling factors based on batch 75 are more representative of the radionuclide distribution in the glass in the plugged exit port.

[5] Only Cs-137 and Sr-90 contribute more than five curies to the residual activity in the melter (WMG 2004b, Table 3).

Table 2. Vitrification Melter Total Activity Estimates (in Curies as of October 1, 2004)

Nuclide	Dose-to-Curie[1]	Concentration x Mass[2]	Nuclide	Dose-to-Curie[1]	Concentration x Mass[2]
Cs-137	4.31E+03	2.12E+03	Am-241	3.00E+00	1.47E+00
Eu-154	1.21E+00	5.90E-01	Am-242m	NA	9.16E-05
Th-228	4.09E-02	2.01E-02	Am-243	3.50E-02	1.72E-02
Th-230	3.65E-04	1.79E-04	Cm-242	7.33E-02	3.60E-02
Th-232	4.01E-04	1.97E-04	Cm-243	1.68E-02	8.24E-03
U-232	5.01E-04	2.46E-02	Cm-244	NA	1.55E-02
U-234	9.81E-03	4.80E-03	Cm-245	NA	1.77E-03
U-235	3.76E-04	1.84E-04			

LEGEND: NA = not available

NOTES: (1) From WMG 2004b, Table 3, as of October 1, 2004 (Table 2-2 of the draft evaluation).

(2) Based on a mass of 326 kg of residual glass in the melter cavity and radionuclide concentrations for the evacuated canister sample analytical data and 99 kg of other glass in the plugged discharge port, considering that the dose rate on canister 266, which was being filled when the discharge port became plugged, was 5.9 times higher than the average dose rate on the two evacuated canisters.

As can be seen from the table, the estimates made by the alternate method are lower, indicating that the dose-to-curie method described in the characterization report (WMG 2004b) produced conservative results. The total activity estimate by the alternative method is approximately 2,240 curies, compared to approximately 4,570 curies by the dose-to-curie in the draft evaluation.

When the two alternative estimates are used to calculate the sums of fractions for Table 1 and Table 2 of 10 CFR 61.55, the results are as follows:

Table 3. Class C Sum of Fractions Comparison

Method	Table 1	Table 2
Original estimate in Table 6-1 of draft evaluation	0.949	0.0459
Table 6-1 revised to add other radionuclides[1]	0.951	0.0459
Alternative method using mass and radionuclide concentrations[2]	0.470	0.0201

NOTES: (1) Pu-242, Am-242m, Cm-245, and Cm-246 (see response to comment WC-2).

(2) Includes Pu-242, Am-242m, Cm-245, and Cm-246.

It is evident from the information just discussed that use of the glass sample scaling factors is appropriate and use of the dose-to-curie estimating method is conservative.

Basis for the Conclusion That There are No Other Significant Sources

The estimating process took into account all glass known to remain inside the vitrification melter in significant amounts, including assuming a one-eighth-inch-thick glass coating on the cavity walls up to a fill height of 28 inches (WMG 2004b). Other surfaces above this level are expected to contain radioactive contamination transported by vapors from the glass pool. Small amounts of glass on the walls and the roof (the underside of the vitrification melter lid) were also reported (Ploetz 2004).

It is possible that the radionuclide distribution in the contamination on the walls and the roof of the vitrification melter could be somewhat different from the radionuclide distribution in the material in

the evacuated canisters. However, the slurry was reasonably homogeneous in this respect as noted previously and the amounts of such contamination would be a small fraction of the residual radioactive material considered in the estimate and, consequently, of negligible importance to the total activity estimate[6].

Likewise, there may be small pockets of residual glass in parts of the vitrification melter not addressed in the estimates and the radionuclide distributions in this material may vary some from radionuclide distribution in the material in the evacuated canisters. The amounts of any such material would be small compared to the amount of residual glass considered in the estimate and thus of negligible consequence.

The residual radioactivity on the external surfaces of the vitrification melter was estimated based on measured surface contamination and determined to be only 0.11 percent of the total activity (WMG 2004b), so this activity is negligible compared to the radioactivity inside the melter.

Conclusions and changes to the draft evaluation: Consideration of this NRC comment and the information provided above has led to the following conclusions:

- The residual radioactivity estimating approach used (WMG 2004b) was technically sound and produced conservative results.
- The scaling factors used were appropriate.
- It would be useful to include information in the evaluation about the results of the alternate estimating method in the context of a crosscheck of the results of the method described in the draft evaluation.

DOE is therefore revising the draft evaluation to include the alternate estimating method results in connection with Table 2-2 and Table 6-1 to demonstrate that the methods used produced conservative results.

In addition, to help place the Class C sum-of-fraction results into context, a footnote will be added to explain that mass of the grout and the volume of the shielded container were not considered in the calculation for conservatism, even though this could have been done consistent with concentration averaging guidance since the grout is required to stabilize the waste package for transportation and disposal purposes.

[6] The 2004 estimate (WMG 2004b) conservatively assumed a continuous 1/8-inch thick layer of dried slurry on melter internal surfaces up to the 48-inch level. If it is assumed that the contamination above this level is contained within a thinner layer of dried slurry the thickness of paint (0.002 inch), then the resulting mass would be less than one percent of the estimated 425 kg mass of hardened glass estimated to remain in the vitrification melter.

Number: IN-2

Subject: The radionuclides included in the inventory estimate from 1987 differ from those in the inventory from 2004.

Basis: NUREG-1854, Section 3.1, advises that the NRC evaluate previous inventory estimates and verify the technical bases for any differences between historical and current estimates (NRC 2007). The 1987 characterization of the sludge and supernatant contains many radionuclides that were screened out in the 2004 inventory in Table 2-2 (e.g. Pu-240, Pu-242, Cm-244, Cm-245, and Cm-246). Additionally, there are a few radionuclides that do appear in the 2004 list, but do not appear in the 1987 list (K-40, Mn-54, Zr-95, Eu-154, Th-228, Th-230, and U-232).

Path Forward: Describe if different screening criteria were used for the 1987 inventory (Eisenstatt, 1986) vs. the 2004 (WMG 2004b) inventory and how the different screening criteria account for the differences in inventories.

DOE response: The differences identified in the comment pertain to the radionuclides addressed and result from the different purposes of the estimates, that is, the radionuclide selection process varied depending on the objectives of the characterization.

1987 Estimates

The 1987 estimates were made for tank waste characterization purposes to identify radionuclides that were present in significant concentrations in the sludge and/or supernatant to support development of pretreatment and treatment plans. A more detailed report of the characterization of the radioactivity in the underground waste tanks (Rykken 1986) included Eu-154, Th-228, Th-230, and U-232. Neither report describes the screening process that resulted in the list of radionuclides reported.

The 1987 estimates were included for context using information from the Eisenstatt report. They do not directly relate to the residual radioactivity inventory in the vitrification melter.

2004 Estimates

The 2004 estimates were made primarily for evaluation of transportation and waste disposal options. They were incorporated into the waste profile documentation (WVES 2010b), which addresses all of the radionuclides mentioned in the NRC comment.

The primary basis for the waste profile documentation radionuclide list is Table E-1 of the Nevada National Security Site waste acceptance criteria document (DOE-NV 2011). This list includes all radionuclides that are of potential importance to disposal of the waste package. Those radionuclides that are present in the waste package in reportable amounts (greater than one percent of the specified action level or greater than one percent of the total activity in the waste) are identified in the waste profile documentation.

Waste Classification Estimates

Table 6-1 of the draft evaluation, which addresses classification of the waste package, includes those radionuclides identified in Table 1 and Table 2 of 10 CFR 61.55 that are present in amounts sufficient to potentially affect the waste classification.

Consistency in Identifying Radionuclides of Interest

As just explained, differences in the radionuclides of interest in the 1987 and 2004 inventories resulted from different characterization objectives. However, evaluation of the NRC comment led to consideration as to whether there should be more consistency between the radionuclides identified as being of interest in the 2004 estimates and the waste profile documentation and those listed for waste classification purposes in Table 6-1. Table 4 below compares radionuclides mentioned in the NRC comment as they have been considered in different estimates.

Table 4. Radionuclides of Interest

Nuclide	$T^{1/2}$ (yr)(1)	1987(2)	2004(3)	Table 6-1	WP	Activity (Ci)(4)
K-40	1.3E+09		x		x	8.19E-02
Mn-54	8.3E-01		x		x	8.57E-02
Zr-95	1.8E-01		x		x	1.65E+00
Eu-154	1.6E+01	x	x		x	1.21E+00
Th-228	1.9E+00	x	x		x	4.09E-02
Th-230	8.0E+04	x	x		x	3.65E-04
Th-232	1.4E+10	x	x		x	4.01E-04
Pu-240	6.6E+03	x	x	x	x	1.21E-01
Pu-242	3.8E+05	x			x	2.28E-05
Cm-244	1.8E+01	x	x	x	x	4.35E-01
Cm-246	5.5E+03	x			x	1.77E-03

LEGEND: $T^{1/2}$ = half-life, WP = waste profile sheet (WVES 2010b)

NOTES: (1) From HEW 1970.
(2) From Rykken 1986.
(3) From Table 3 of WMG 2004b and Table 2-2 of the draft evaluation.
(4) From Table 3 of WMG 2004b, except for Pu-242 and Cm-246, which were scaled from Pu-240 and Cm-244, respectively, using data from the waste profile documentation (WVES 2010b).

The basis for some radionuclides not being listed in Table 6-1 amounts to low activity in the waste, short half-life, or both, which should be clear from consideration of half-life and activity information in Table 1.

Comment WC-2, which is addressed below, raises a related issue pertaining to Table 6-1 and the need to include Am-242m and Cm-245 in the table. For the reasons discussed in that comment, DOE is changing Table 6-1 to include Pu-242, Am-242m and Cm-245.

Conclusions and changes to the draft evaluation: More consistency in the radionuclides identified in the waste profile documentation and in the tables of the draft would improve clarity. To this end, Table 2-2 and Table 6-1 are being changed to add Pu-242, Am-242m, Cm-245, and Cm-246. The other non-transuranic radionuclides K-40, Mn-54, Zr-95, Eu-154, Th-228, Th-230, and Th-232 would have no effect on the waste classification and therefore are not being added to Table 6-1.

Number: IN-3

Subject: It is unclear if DQOs were established for the analysis supporting waste characterization (WMG 2004b).

Basis: NUREG-1854, Section 3, guides the NRC to review data quality assessments and verify if applicable data quality objectives were met.

Path Forward: DOE should indicate if DQOs were established for the characterization of the inventory. If they were not, DOE should discuss the data quality assurances that took place.

> **DOE response:** Data quality objectives were not established for the characterization of the inventory. However, appropriate quality assurance protocols were specifically followed to validate the dose rate data and the sample analytical data used in the characterization (Lachapelle 2003).
>
> These data, included those from the glass shard samples, were validated pursuant to the requirements of the *Characterization Management Plan for the Facility Characterization Project* (Michalczak 2004). This plan provides data quality objectives and describes how the data were validated. A copy of the work order used to collect the dose rate data used in characterization of the vitrification melter appears in the radioisotope inventory report (Lachapelle 2003).
>
> The WMG characterization was also independently reviewed by the site contractor, West Valley Nuclear Services Company (WVNSCO), and changes from this review were incorporated.

Conclusions and changes to the draft evaluation: The process used to make the estimate was technically sound. No changes to the draft evaluation related to DQOs for the characterization are planned.

Number: IN-4

Subject: Uncertainty accounted for in the calculation of the waste inventory scaling factors is difficult to interpret.

Basis: NUREG-1854, Section 3.1, states that NRC should verify that analytical uncertainties are either propagated into calculations of waste inventory or have been adequately bounded (NRC 2007). A prior waste characterization states that the analytical uncertainty for the measurements of Cs-137 is about 3-5%, the uncertainty around thorium is about 20%, and the uncertainty for the remaining actinides is 50% (Eisenstatt, 1986). These estimates match up with the uncertainty ranges provided for the glass samples (WMG 2004b).

Path Forward: Describe if uncertainty was taken into account in determining the scaling factors, or provide a basis for why it is not important to take into account the uncertainties. Describe the technical basis for using the average of the samples instead of a lower confidence limit (for Cs-137) or upper confidence limit (for Th-232 or actinides) for determining scaling factors.

> **DOE response:** The uncertainty in the glass sample analytical data used to determine the activity scaling factors was not directly taken into account. However, this uncertainty was small and was bounded by the conservatism inherent in the dose-to-curie estimating method (WMG 2004b) used to make the residual radioactivity estimates and determine the waste classification.
>
> Please note that the uncertainty in the glass sample analytical data was smaller than the uncertainly in the early analytical data in the Eisenstatt report that is mentioned in the basis. For example, the uncertainty in concentrations of Cs-137 and Sr-90 that dominate the activity estimate was less than three percent. The uncertainty in concentrations of Pu-238 and Am-241 that control waste classification was approximately six percent and five percent, respectively[7].
>
> The response to comment IN-1 demonstrates that the residual radioactivity estimating method used (WMG 2004b) was inherently conservative. As noted previously, the total activity estimate by the alternative method was approximately 2,240 curies, compared to approximately 4,570 curies by the dose-to-curie methods used in the draft evaluation.
>
> Uncertainty in the sample analytical data was not used in calculating the activity scaling factors because multiple results were available and they were consistent. Geometric averaging of scaling factors is a common practice throughout the commercial nuclear power industry in cases where more than one representative sample is available. This practice is incorporated into the Radman™ software used at West Valley, which has been reviewed and approved by the NRC (NRC 2011b).
>
> This practice is also consistent with NUREG/CR-6567/PNNL-11659, *Low-Level Waste Classification, Characterization, and Assessment: Waste Streams and Neutron-Activated Metals* (NRC 2000), which states that it "is important that waste generators utilize the most accurate scaling factors possible, so that reliable estimates of these nuclides [ones for which activity scaling actors are used] can be made."

Conclusions and changes to the draft evaluation: The use of average values without regard for uncertainty in the analytical data to develop activity scaling factors is consistent with accepted practice and the conservatism in the estimating process bounds the uncertainty in the analytical data. No changes to the draft evaluation in this area are planned.

[7] These uncertainty values are based on the uncertainty in the glass sample analytical data as shown in the laboratory reports, which are included in the characterization report (WMG 2004b).

Number: IN-5

Subject: There is not sufficient technical basis for the volume of residual material remaining in the Melter, and the draft WIR evaluation provides conflicting assumptions regarding the amount of residual material.

Basis: NUREG-1854, Section 3.1, states that NRC staff should confirm that DOE has adequate technical basis for estimating the volume of the waste, and that the reviewer should compare the results of calculated estimates with sampled values to assess the reliability of the estimated values (NRC 2007).

Page 23 of the draft WIR evaluation states, "…approximately 300 kg of glass remained in the Melter cavity, a heel of approximately eight inches. This estimate was based on a combination of level detector responses, preliminary canister weights, and thermocouple responses during the evacuation evolution (Lachapelle, 2003)". Page 24 of the draft WIR evaluation states, "A model prepared…to represent the complex geometry of the Vitrification Melter cavity. This model accounted for a layer of contamination 0.125-inch thick on the cavity surfaces to its full height of 28 inches, as well as the eight inches of solidified glass in the bottom of the cavity. The basis for the thickness of the contamination layer is previous WVDP experience in Vitrification Melter testing described in the Slurry Fed Ceramic Melter Disassembly Report (Brooks, 1993)." However, it is unclear how a value of 8 inches was determined from the information contained in Lachapelle and Brooks references.

Furthermore, the assumption of 8 inches and 300 kg does not seem to match the activity removal estimates and concentrations (WMG 2004b). The gamma measurements indicated there are 4,120 Ci in the Melter (page 38), then simply dividing this amount by 300 kg, and adjusting units gives you a Cs-137 concentration of 1.37E4 µCi/g, which is larger than the 2.0E3 µCi/g concentration that was measured in the glass in the evacuated canisters. Similarly, if 300 kg is assumed with a density of 2.5 g/cm3 and a Cs-137 concentration of 5.0E3 µCi/cm3 (page 38), then this suggests only 600 Ci of Cs-137 remains in the tank.

Path Forward: (1) Provide an explanation as to why the assumption provided about the mass and volume of remaining material do not seem to match the activity content that was measured using the gamma surveys. (2) Provide any additional information supporting the technical basis for the amount of residual waste in the Melter. (3) Provide the references L. L. Petkus, Senior Engineer, High-Level Waste Completion Project, to Distribution, "Glass in Melter," dated September 9, 2002 and "Application of the Evacuated Canister System for Removing Residual Molten Glass from the West Valley Demonstration Project High-Level Waste Melter," J. J. May et al., Waste Management 2003 Conference.

NRC request (1): Provide an explanation as to why the assumption provided about the mass and volume of remaining material do not seem to match the curie content that was measured using the gamma surveys.

> **DOE response:** The dose-to-curie estimating method (WMG 2004b) produced conservative results compared to estimates made by multiplying radionuclide concentrations by the mass of the remaining glass as shown in the response to comment IN-1. The estimates made by the latter method are considered to be accurate with a relatively small uncertainty band[8].

NRC request (2): Provide any additional information supporting the technical basis for the amount of residual waste in the Melter.

[7]As noted previously, the uncertainty in concentrations of Cs-137 and Sr-90 that dominate the activity estimate was less than three percent. The uncertainty in concentrations of Pu-238 and Am-241 that control waste classification was approximately six percent and five percent, respectively.

DOE response: The Petkus memo (Petkus 2002b) simply states "Based on the sludge probe, canister weight (preliminary) and the thermocouple responses during the evolution, there is about 300 kg of glass left in the vitrification melter or 8 inches. The melter started with 26 inches of level."

Information from the inspections of the dismantled test melter (Brooks 1993) was used in establishing the 0.125-inch assumed glass thickness on the vitrification melter internal surfaces.

An independent estimate of the remaining glass in the vitrification melter cavity has been made as described in Attachment 2. This estimate, which considered actual vitrification melter level data and measured canister weights, produced an estimate of a heel of hardened glass approximately 6.5 inches deep, which substantiates the conclusion that the eight-inch value used in the activity estimate is conservative.

NRC request (3): Provide the references L. L. Petkus, Senior Engineer, High-Level Waste Completion Project, to Distribution, "Glass in Melter," dated September 9, 2002 and "Application of the Evacuated Canister System for Removing Residual Molten Glass from the West Valley Demonstration Project High-Level Waste Melter," J. J. May et al., Waste Management 2003 Conference.

DOE response: These references are being provided to NRC with the comment responses, along with other listed references that were not provided to NRC previously.

Conclusions and changes to the draft evaluation: The volume estimate of 300 kg based on an eight-inch depth of residual glass in the vitrification melter cavity is conservative. Additional information supporting this statement will be included in the evaluation, including a brief summary of Attachment 2.

Number: KR-1

Subject: The basis for the different radionuclides listed in the Melter inventory in Table 2-2 and the list of key radionuclides in Table 4-3 are not explained.

Basis: NUREG-1854, Section 3.2, states that the list of key radionuclides is expected to be specific to a particular waste determination, and that DOE may start with the radionuclide inventories and eliminate radionuclides from the list of potentially key radionuclides based on screening criteria (NRC 2007). The reviewer should assess the reasonableness of any screening criteria used to remove radionuclides from the list of potential key radionuclides. The 2004 inventory of the waste provided in Table 2-2 of the draft WIR evaluation contains many radionuclides that were not included as key radionuclides in Table 4-3. For example the following transuranics with half-lives greater than five years are listed in the inventory of the sludge and supernatant predicted for 1987 (Eisenstatt, 1986), but they are not listed in Table 2-2 (derived from reference (WMG 2004b)) nor are they listed in Table 4-3 of the draft WIR evaluation (half-lives in parenthesis): Am-242m (152 yr), Cm-245 (8,500 yr), Cm-246 (4,730 year). Cm-245 and Cm-246 are listed in the Waste Profile Sheet as "reportable" radionuclides, but do not appear in the list of key radionuclides (WVES 2010b). If these radionuclides are unimportant to risk, DOE should describe the screening criteria used to make this determination.

Path Forward: Please provide any screening criteria (e.g., short half-life, low activity level) used to eliminate the radionuclides that appear on the inventories provided for the Melter, but do not appear in the key radionuclides list in Table 4-3 of the draft WIR evaluation?

DOE response: Low total activity in the waste was the basis for omitting Am-242m, Cm-245, and Cm-246. That is, they are not present in the waste package in amounts sufficient to be important to disposal site performance and therefore are not included among the key radionuclides.

The three radionuclides are not among those important to the performance assessment of Area 5 at the Nevada National Security Site. Table E-1 of the Nevada National Security Site waste acceptance criteria document (DOE-NV 2011) lists action levels for a large suite of radionuclides; those with concentrations in the waste exceeding one percent of these values or one percent of the total waste package activity must be reported on the waste profile sheet. Table 5 compares concentrations of these radionuclides in the melter waste package with the action level in Table E-1, except for Am-242m, which is not listed in that table.

Table 5. Additional Alpha-Emitting Radionuclides

Radionuclide	Ci	Bq/m^3	Action Level (Bq/m^3) (1)	Basis
Am-242m	1.88E-04	3.76E+08	NA	Note (2).
Cm-245	9.56E-05	1.92E+07	4.6E+10	Note (3).
Cm-246	1.56E-05	3.12E+06	9.2E+10	Note (3).

LEGEND: NA = not available.

NOTES: (1) From Table E-1 of DOE-NV 2011 .

(2) From Rykken Table 22, ratio to measured Am-241 from WVES 2010b, conservatively assuming that alpha radiation represents 0.5 percent of emissions.

(3) WVES 2010b, Table 4-1 of Technical Basis Document, concentration based on 425 kg mass.

Table 5 shows Cm-245 and Cm-246 to be less than one percent of the action level and therefore not reportable. While no action level is specified for Am-242m, the concentration of Am-242m is

less than one percent of the action levels for the two americium radionuclides listed in Table E-1, which are Am 241 (1.7E+11 Bq/m^3) and Am-243 (5.8E+10 Bq/m^3).

The Waste Control Specialists low-level waste (LLW) disposal facility waste acceptance criteria (WCS 2008) does not have radionuclide action levels like the Nevada National Security Site and Am-242m, Cm-245, and Cm-246 are not among the radionuclides important to the Waste Control Specialists performance assessment (WCS 2007). Consequently, the subject radionuclides are not key radionuclides for disposal at the Waste Control Specialist facility.

Conclusions and changes to the draft evaluation: It would be useful for clarity to address this matter in the evaluation. Information is being added to Section 4.1.6 of the draft evaluation to this end.

Number: KR-2

Subject: The list of key radionuclides does not appear to be based on those radionuclides from the Melter inventory that contribute most significantly to risk to the public, workers, and the environment.

Basis: NUREG-1854, Section 3.2.2, states that the reviewer should ensure that the list of key radionuclides does not omit radionuclides that may be predicted to cause a significant contribution to risk. The reviewer should also evaluate the process DOE used to identify those radionuclides that contribute most to dose, and identify the uncertainties that are expected to have the most significant effect on predicted dose (NRC 2007). The key radionuclides identified in the draft WIR evaluation for the Melter appear to be primarily based on the radionuclides listed in 10 CFR 61.55 and the performance assessments for the disposal facilities as opposed to identifying radionuclides from the actual Melter inventory that most significantly contribute to risk. Because the residual Melter waste is derived from HLW, it is possible that the Melter inventory contains radionuclides in quantities (i.e., activity and concentration) that exceed those estimated for commercial low-level radioactive waste streams in the technical basis to develop the Section 61.55 waste classification tables (NUREG-0945) and/or the performance assessments for the potential LLW disposal facilities.

Path Forward: (1) Indicate those radionuclides contributing most significantly to risk to members of the public, including inadvertent intruders, workers, and the environment based on the radionuclide inventory derived specifically for the West Valley Melter considering uncertainty in the timing and magnitude of peak dose at a potential disposal facility. (2) Indicate why DOE is confident that no risk-significant radionuclides were omitted from the list of key radionuclides provided in Table 4-3.

NRC request (1): Indicate those radionuclides contributing most significantly to risk to members of the public, including inadvertent intruders, workers, and the environment based on the radionuclide inventory derived specifically for the West Valley vitrification melter considering uncertainty in the timing and magnitude of peak dose at a potential disposal facility.

DOE response: This comment is addressed by discussing the following matters:

- The radionuclides present in the vitrification melter and their estimated activities,
- The radionuclides present in the vitrification melter that are important to the performance assessments and waste acceptance requirements of the two LLW disposal facilities with respect to member of the public and the environment,
- The basis for selection of these radionuclides from the standpoint of members of the public (excluding inadvertent intruders) and the environment,
- The basis for selection of these radionuclides from the standpoint of inadvertent intruders, and
- The radionuclides that contribute most to worker dose.

Radionuclides Present in the Vitrification Melter

Table 2-2 of the draft evaluation identifies the radionuclides present in the vitrification melter and their estimated activities. Table 2 above includes several other radionuclides not listed in Table 2-2 and their activities estimated by an alternate method. The response to comment IN-2 indicates that the other radionuclides are being added to Table 2-2.

Radionuclides Important to Performance Assessments and Waste Acceptance Criteria

Radionuclides that are important to the performance assessments and waste acceptance criteria are identified in Table 6 below."

Table 6. Key Radionuclide Summary (Members of the Public)

Nuclide	NNSS				WCS			
	Important to PA(1)	WAC Action Level (Bq/m^3)(2)	Melter (Bq/m^3)(3)	Melter % of Action Level	Important to PA(1)	License Limit (Ci)(4)	Melter (Ci)(5)	Melter % of Limit
C-14					X	1.8E+02	2.12E-02	0.012
Tc-99	X	3.2E+09	1.94E+07	0.61	X	3.5E+01	1.11E-02	0.012
I-129					X	1.5E-01	5.64E-03	3.8(5)
Th-229	X	2.8E+10	2.8E+02	<0.0001				
U-233	X	8.3E+10	3.59E+07	4.3				
U-234	X	1.3E+10	1.71E+07	0.13				
U-238	X	3.5E+11	3.93E+06	0.001				

LEGEND: NNSS= Nevada National Security Site, PA = performance assessment. WAC = waste acceptance criteria, WCS = Waste Control Specialists

NOTES: (1) From Table 4-3 of draft evaluation.
(2) From waste acceptance criteria (DOE-NV 2011).
(3) From Table 5-3 of draft evaluation.
(4) From WCS license (TCEQ 2010).
(5) From Table 2-2 of draft evaluation or NNSS waste profile documentation (WVES 2010b), except for I-129, which his from Table 6-1 of the draft evaluation. I-129 did not exceed the minimum detectable activity in the glass samples, so the 5.64E-02 Ci estimate is based on the detection limit (WMG 2004b, Table 3).

This table shows that the key radionuclides in the vitrification melter waste package with respect to members of the public, that is, those in the "Important to PA" columns, are C-14, Tc-99, I-129, Th-229, U-233, U-234, and U-238, with the caveat that I-129 was not detectable in the samples taken from the evacuated canisters.

Member of the Public (Excluding Inadvertent Intruders) and the Environment

As shown in Table 6, the radionuclides in the vitrification melter waste package that would contribute most significantly to long-term risk to members of the public and to the environment from disposal at the Nevada National Security Site are Tc-99 and U-234, although the associated risks would be negligible. This conclusion is based on the following information:

- The limiting pathway-scenario combination for a member of the public from the Area 5 Radioactive Waste Management Site is the all-pathways, resident farmer (NST 2010).
- The special performance assessment performed to evaluate disposal of the vitrification melter showed that the maximum all-pathways dose to a member of the public for the resident farmer scenario is predominately from Tc-99 (78 percent) and Pb-210 (13 percent), with the predominant source of Pb-210 being U-234 (DOE 2010b).
- The concentrations of Tc-99 and U-234 in the vitrification melter waste package are less than one percent of the waste acceptance criteria radionuclide action levels and less than one percent of the total waste package activity (WVES 2010b).

- Analysis shows that the impact of disposal of the vitrification melter in the Area 5 Radioactive Waste Management Site to a member of the public through the all-pathways resident farmer scenario would be negligible (DOE 2010b).

The radionuclides in the vitrification melter waste package that would contribute most significantly to long-term risk to members of the public and the environment from disposal at the Waste Control Specialists LLW disposal facility are Tc-99 and I-129[9] because these radionuclides are predicted to produce essentially 100 percent of annual exposure to a member of the public, as indicated in footnote 38 on page 53 of the draft evaluation.

Inadvertent Intruders

The radionuclides in the vitrification melter waste package that would contribute the most risk to an inadvertent intruder at the Nevada National Security Site are Th-229, U-223, U-234, and U-238, although the associated risks would be negligible. This conclusion is based on the following information:

- The limiting acute intruder scenario for a member of the public at the Area 5 Radioactive Waste Management Site is the construction scenario (DOE 2010b).
- The special performance assessment showed that the maximum dose to a member of the public for this scenario is predominately from Th-229 (30 percent), U-233 (16 percent), U-234 (10 percent), and U-238 (29 percent) (DOE 2010b).
- Analysis shows that the impact of disposal of the vitrification melter in the Area 5 Radioactive Waste Management Site to a member of the public through the all-pathways resident farmer scenario would be negligible (DOE 2010b).

All four of these radionuclides are identified as key radionuclides in Table 4-3 of the draft evaluation. However, Th-229 does not appear in Table 2-2 of the draft evaluation which shows the vitrification melter total activity estimates because it is evident that the Th-229 inventory is insignificant[10].

Information on the radionuclides that would contribute the most risk to risk to the inadvertent intruder for the Waste Control Specialists LLW disposal facility is not available.

Dose to Workers

As the dominant gamma-emitting radionuclide in the vitrification melter, Cs-137 would contribute the most risk to workers at the WVDP and at the waste disposal site. Preparation of the waste package for shipment by DOE would involve potential worker risks from other radionuclides though indirect exposure pathways. However, compliance with the WVDP radiological control program requirements will ensure that such exposure would be insignificant.

In Summary

Table 4-3 of the draft evaluation includes other key radionuclides based on their inclusion in Table 1 or Table 2 of 10 CFR 61.55. The radionuclides listed in Table 4-3, along with these other radionuclides, comprise all of the risk-significant radionuclides in the vitrification melter waste

[9] As noted previously, I-129 is not present in the waste package above the detection limit.
[10] The scaling factors used in the estimates of residual activity in the vitrification melter are based on analytical data from the two glass shard samples from the evacuated canisters. These data show that Th-229 in these samples was undetectable, that is, it did not exceed the minimum detectable activity in either sample. (WMG 2004b)

package for the reasons just discussed. The melter inventory does not contain radionuclides in quantities that exceed those estimated for commercial low-level radioactive waste streams in the technical basis to develop the Section 61.55 waste classification tables and/or the performance assessments for the potential LLW disposal facilities.

NRC request (2): Indicate why DOE is confident that no risk-significant radionuclides were omitted from the list of key radionuclides provided in Table 4-3.

DOE response: DOE is confident that no risk-significant radionuclides were omitted from the list of key radionuclides for the reasons discussed in connection with NRC request (1).

Conclusions and changes to the draft evaluation: It would be useful for clarity to include I-129 and Th-229 among the radionuclides in the Table 2-2 estimates. A note will be added to the table indicating that the amounts of these radionuclides present in the waste package is insignificant based on sample analytical data used in the estimates. A correction will be made to Table 5-4 of the draft evaluation to show the I-129 values in Table 6 above.

Number: MEP-1

Subject: Additional technical basis should be provided to support the assumption regarding removal of radionuclides being completed in equal proportions.

Basis: NUREG-1854, Section 3.1, states that the reviewer should consider whether the results of a comparison made for one radionuclide can provide information about the expected accuracy of the estimated inventory of other radionuclides (NRC 2007). Page 39 of the draft WIR evaluation states that, "other key radionuclides were removed from the Vitrification Melter with approximately the same efficiency as Cs-137." The reasons provided are that the waste was generally homogeneous, but there is not a discussion about chemistry involved in the flushing process, and whether or not the nitric acid or demineralized water would be expected to chemically remove Cs-137 or other key radionuclides to a greater extent than others.

Path Forward: DOE should provide additional technical basis for why flushing would be expected to remove radionuclides in equal proportions.

DOE response: As discussed in the response to comment MEP-4, the vitrification melter was not flushed in the traditional sense. The flush solutions entered the vitrification melter where they were homogenized with the molten glass, reducing the radionuclide concentrations of the material, and the molten glass would be air lifted to fill a HLW canister. This process continued as the last group of canisters was produced. With this process, the nitric acid and demineralized water would immediately lose their identity in the molten glass pool and have no independent impact on the vitrification melter surfaces and would not selectively remove particular elements or radionuclides.

One measure of consistency in removal of different radionuclides by the system flushes on the other vitrification equipment could be the differences between radionuclide distributions in waste being fed to the vitrification melter and radionuclide distributions in the final flush solutions processed through the melter. Major differences in fractional abundance in such a before and after flushing comparison could be indicative of one or more radionuclides plating out on equipment or piping surfaces or otherwise being removed to a lesser extent than other radionuclides.

To evaluate whether such a comparison would be meaningful, Table 7 compares the before and after flushing radionuclide abundance factors of representative radionuclides using two sets of data:

- Analytical data from samples of slurry batch 75, which was made up of the last material removed from Tank 8D-2, and
- Analytical data from samples of the decontamination flush solutions removed from the vitrification melter into the evacuated canisters, that is, the data set from analysis of the glass shard samples taken from the evacuated canisters.

Table 7. Radionuclide Fractional Abundance Comparison

Radionuclide	Fractional Abundance Based on CFMT Batch 75 Data(1)	Fractional Abundance Based on Evacuated Canister Data(2)
Co-60	2.13E-05	1.95E-05
Sr-90	6.94E-02	5.38E-02
Tc-99	1.27E-05	2.37E-06
Cs-137	9.30E-01	9.38E-01
Eu-154	2.14E-04	2.73E-04

Table 7. Radionuclide Fractional Abundance Comparison

Radionuclide	Fractional Abundance Based on CFMT Batch 75 Data[1]	Fractional Abundance Based on Evacuated Canister Data[2]
Np-237	5.70E-07	1.33E-06
Pu-238	1.01E-04	1.47E-04
Am-241	3.08E-04	6.42E-04

LEGEND: CFMT = concentrator feed makeup tank

NOTES: (1) Based on VAST sample 01-2498 (WVNSCO 2002a).
(2) From WMG 2004b, Table 3.

Table 7 shows that the fractional abundance before and after flushing to be similar for most radionuclides. The Cs-137 fractional abundance difference is insignificant and the Sr-90 difference is small. The largest difference is for Tc-99. In general, these data support the assertion that the flushing removed radionuclides in approximately equal proportions.

Conclusions and changes to the draft evaluation: It would be useful for clarity to provide supporting information in the draft evaluation for the assertion that "Since flushing removed molten glass, all radionuclides in the mixture were removed in approximately equal proportions." To this end, information will be included explaining how flushing impacted the vitrification melter. Note that the section in the draft evaluation on removal of key radionuclides will be revised for other reasons as well as discussed below.

Number: MEP-2

Subject: The assessment of the amount of radioactivity present prior to the flushing of the Melter is unclear.

Basis: NUREG-1854, Section 3.3.2, states that the reviewer should verify that reported removal efficiencies are reasonably reliable. Page 38 of the draft WIR evaluation states a typical Cs-137 concentration of 3.0e4 µCi/cm3 was assumed in the glass prior to the first flushing. The draft WIR evaluation also assumes a typical level in the Melter of 26 inches. However, a technical basis for the typical values is not provided. The report provides a typical estimate for a glass canister in 1990, which, decayed to 2004, is 2.5E4 µCi/cm3, but this value does not match the typical value assumed (Eisenstatt, 1986). Also, on page 37 of the draft WIR evaluation, it is stated that towards the end of the campaign, the liquids being retrieved from the tanks were increasingly dilute. This would imply that a typical concentration may not adequately represent the Cs-137 concentration in the glass formed towards the end of the campaign.

Path Forward: Provide additional technical basis for the assumptions regarding the starting activity of 30,000 Ci of Cs-137 in the Melter prior to flushing. Clarify if the 30,000 Ci includes the activity in the plugged discharge port or not.

DOE response: The basis for each key assumption was as follows:

- The working level of the vitrification melter is 26 inches from the bottom of the melter cavity based on the design of the melter and the molten glass exit paths.[11]
- The vitrification melter was full to its 26-inch working level, based on operating band requirements specified in the standard operating procedure (WVNSCO 2002c) that were noted previously.
- The assumed volume of material in the vitrification melter (1000 L) was an approximate value. The volume based on the working capacity of 227 gallons (Vance, et al. 1997) equates to 860L or 8.60E+05 cm^3.
- The assumed value of 3.0E+04 µCi/cm^3 Cs-137 in the waste was a reasonable value based on analytical data from samples taken from the concentrator feed makeup tank. For example, the 1.16E+04 µCi/g Cs-137 concentration for batch 75 (WVNSCO 2002a) equates to 2.9E+04 µCi/cm^3.[12]

The 30,000 Ci estimate is the product of the 1000 L volume and the 3.0E+04 µCi/cm^3 Cs-137 concentration. The activity in the plugged discharge chamber was not included since the discharge chambers do not normally contain glass except when it is being poured into a canister. Although the 1000 L assumed volume was approximately 16 percent greater than the working capacity of the melter, the volume of hardened glass in the plugged discharge chamber was estimated to be approximately 99 kg out of a total estimated mass of 425 kg, that is, approximately 23 percent of the hardened glass remaining in the vitrification melter.

[11] The top of the exit up-ramp is 26 inches from the bottom of the melter cavity. Figure 2-7 of the draft evaluation shows the exit up-ramp, which can be seen to be at approximately the midpoint of the 4.5-foot high melter cavity. The tube used inject air to "lift" the molten glass over the top of the ramp can also be seen in the figure. If the melter cavity were filled with, for example, 26-inch deep water, the water surface would be located at the top of the ramp. If more liquid were to be added, it would flow down the trough and keep the level in the melter at 26 inches.

[12] The reference to the Eisenstatt 1986 report in the NRC comment basis is not germane to the issue.

Conclusions and changes to the draft evaluation: The approach used to make the 30,000 Ci estimate did not include the material in the plugged discharge chamber, although the estimate included significant conservatism. It was appropriate not to include the activity in the plugged discharge chamber because the starting activity was intended to serve as a baseline for demonstrating the effectiveness of the melter key radionuclide removal efforts, which were not designed to remove hardened glass from the plugged discharge chamber.

Changes are being made to the draft evaluation regarding the effectiveness of the flushing, including the estimated initial activity, as described in the response to comment MEP-4.

Number: MEP-3

Subject: The values provided for the remaining radioactivity are inconsistent.

Basis: The draft WIR evaluation, page 25, states that there were 4,062 Ci of Cs-137 in the cavity and 252 Ci of Cs-137 in the plugged discharge tube, for a total of 4,314 Ci of Cs-137 (WMG 2004b). This value does not match the value presented on page 38, which states that after the first three flushes and use of evacuated canisters, the Melter contained approximately 4,120 Ci of Cs-137 in the cavity and approximately 540 Ci "ex-cavity", for a total of 4,660 Ci of Cs-137 It seems like this number is from Revision 1 (WMG 2004a), not Revision 3. Revision 1 was not provided, but it is cited (Perdue 2004).

Path Forward: DOE should explain the reason for this inconsistency.

> **DOE response:** The comment is correct in that the 4,660 Ci and the related values are from Rev 1 of the WMG report (WMG 2004a in this document).

Conclusions and changes to the draft evaluation: This portion of the draft evaluation is being revised as explained in the response to comment MEP-4.

Number: MEP-4

Subject: The basis for the calculation of the remaining Melter inventory after flushing and the remaining inventory after evacuated canisters is unclear.

Basis: NUREG-1854, Section 3.3.2, states that the reviewer should verify that reported removal efficiencies are reasonably reliable (NRC 2007). The basis for the 4,460 Ci remaining after the third flush is provided (WMG 2004b), but it is unclear how the other numbers were derived. Page 38 of the draft WIR evaluation states that, "…8,600 curies were remaining in the Melter after flushing but before Evacuated Canister System removal." This number is derived by assuming a 30,000 Ci starting point, and a 4,120 Ci ending point to calculate that 25,900 Ci were removed by both efforts. Knowing the volume and concentration of the glass removed by the evacuated canisters, it is determined that 4,500 Ci were removed by the canisters, and 21,400 Ci were removed by the flushing alone. Thus, 30,000 – 21,400 or 8,600 Ci remained in the Melter after flushing alone. However, Table 4-4 lists 8,668.91 Ci remaining in the Melter after the second flush. It appears the way the values are presented in Table 4-4 is in disagreement with the logic presented on the page. Also, if 8,600 Ci remained after flushing, the volume in the Melter remaining after flushing can be derived by dividing by the concentration assumed to be in the glass after flushing and use of the canisters of 5.0e3 µCi/cm3, which equates to 1,720 Liters. Since this volume is larger than the 1,000 Liters assumed at the start, it might suggest that glass was added to the Melter cavity as a result of flushing the rest of the system. However, the possibility of adding glass to the Melter cavity during flushing, and how this impacts the removal efficiency calculations for flushing is not discussed.

Path Forward: (1) Provide a description of how the values of 16,126.60 Ci and 8,668.91 Ci in Table 4-4 were determined, and if these values include the activity assumed to be only in the cavity, or also include activity remaining in the plugged discharge port. (2) Clarify if 8,668 Ci remained after the second or third flush (prior to evacuated canisters). (3) Describe if any measurements were taken between flushes to support assumptions regarding the efficiency of each individual flush. (4) Describe the volume of decontamination fluids used to flush material through the system. (5) Describe if any volume of glass was expected to be added to the cavity during the flushing process from the process lines, or tanks associated with the Vitrification Facility process. (6) Describe if there is a portion of the cavity that cannot be readily discharged via flushing (e.g., stagnant portion on the bottom). (7) Describe the uncertainty around the estimated 4,460 Ci remaining (e.g., from the uncertainty in the detection instrument used, or the uncertainty in the assumptions about remaining geometry).

DOE response: Recognizing that the melter had a limited life expectancy, DOE developed a deliberate decision-making process to evaluate the optimal timing of melter shutdown. This process made use of a systems approach and considered a number of factors including the waste-incidental-to-reprocessing criteria of DOE Manual 435.1-1 as they applied to the HLW retrieval and vitrification system. In late 2000, DOE commissioned a Vitrification Completion Team, consisting of representatives from DOE, the New York State Energy Research and Development Authority, and West Valley Nuclear Services, as well as NRC to review issues surrounding the ability to complete vitrification operations (VCT 2001). The Vitrification Completion Team developed an approach to retrieving waste from the tanks, washing and characterizing the residual tank materials and flushing the vitrification system prior to completing a controlled shutdown of the melter.

The vitrification system flushes involved flushing Tank 8D-1, Tank 8D-2, the waste header, Tank 8D-4, sludge mobilization system piping, portions of the supernatant treatment system, the low-level waste evaporator, the concentrator feed makeup tank, the melter feed hold tank, the submerged bed scrubber,

and the vitrification melter. The concentrator feed makeup tank and the melter feed hold tank were required to be flushed by two passes using high-pressure spray. For the vitrification melter, this process involved slowly feeding the decontamination (flushing) material to the melter while periodically airlifting molten glass into a waiting canister, thereby removing key radionuclides from the melter and reducing its radionuclide inventory[13]. (WVNSCO 2002b)

The flushing plan (WVNSCO 2002b) acknowledged concerns over limited vitrification melter life but stated that "flushes may be repeated based on the results obtained." Based on this criterion, the melter feed hold tank was flushed three times instead of the required two times. A deliberate decision was made that the flushing had accomplished its objectives and reached the point of diminishing returns before the flushing was concluded.

A detailed description of the flushing accomplishments appears in the *Report on Deployment of Miscellaneous Tanks and Piping Cleaning Equipment and Methodology* (WVNSCO 2002g). This report includes photographs that demonstrate the effectiveness of the flushing on the internals of the concentrator feed makeup tank and the melter feed hold tank. Additional flushing of these vessels and other vitrification system equipment was determined to be unnecessary based on the visual inspections that showed the vessel interiors to be clean and free of visible waste, and based on dose rate and radionuclide concentration data.

NRC request (1): Provide a description of how the values of 16,126.60 Ci and 8,668.91 Ci were in Table 4-4 were determined, and if these values include the activity assumed to be only in the cavity, or also include activity remaining in the plugged discharge port.

> **DOE response:** The 16,126.60 Ci estimate was based on the 1000 L volume of material in the vitrification melter cavity and the measured Cs-137 concentrations in batch 76, using a material density of 2.6 g/cm^3. The basis for the 8,668.91 Ci estimate could not be determined, but this value is clearly conservative as discussed below. These estimates did not include the activity in the plugged discharge chamber since they were based on the vitrification melter working volume.

NRC request (2): Clarify if 8,668 Ci remained after the 2nd or 3rd flush (prior to evacuated canisters).

> **DOE response:** The 8,668 Ci estimate applies after completion of processing the second batch of decontamination (flush) solution from the concentrator feed makeup tank and the melter feed hold tank and prior to use of the evacuated canisters. For clarification, the melter was flushed twice – that is, two batches of decontamination solutions were processed in the melter – and not three times.
>
> Evaluation of this request has led to the conclusion that estimates of flushing effectiveness given in Table 4-4 of the draft evaluation and Table 11 of the Perdue report on which Table 4-4 was based are conservative. This conclusion was reached after consideration of several related matters:
>
> - The difficulty and inherent inaccuracy in correlating vitrification melter feed batch radionuclide concentrations with the radionuclide concentrations in the melter at a particular time,
> - How decontamination flush solutions impacted the vitrification melter, and
> - Estimates made by alternative methods are lower than the 8,668 Ci estimate in Table 4-4.

[13] This process amounted to a feed-and-bleed process with slow continuous feed and periodic bleeding of material from the melter cavity.

Correlating Cs-137 Concentration in Feed Material Batches and Canister Dose Rates

The vitrification melter design did not include provisions for collecting samples from the material actually inside the melter. The best available data on radionuclide concentrations in the vitrification melter glass pool over the course of the vitrification program are the analytical data from samples drawn from the concentrator feed makeup tank (WVES 2011), although analytical data from random sampling of glass in 28 HLW canisters were also considered (WVES 2010a).

As explained in the response to comment IN-1, the composition of the glass pool changed slowly over time as additional batches were added to the vitrification melter and molten glass was poured into the HLW canisters. Because of this slow rate of change, the radionuclide concentrations in samples of material in the concentrator feed makeup tank cannot be directly correlated with the radionuclide concentrations in the melter at a specific time. That is, the vitrification melter glass pool could have contained material from two or more concentrator feed makeup tank batches at the same time.

As indicated previously, the data from analysis of the evacuated canister glass shard samples provide definitive information on the radionuclides concentrations of the material that remains in the vitrification melter cavity.

Impact of Decontamination Solutions on the Vitrification Melter

The discussion in Section 4.2.3 of the draft evaluation and the Perdue report implies that the vitrification melter internals were flushed with decontamination solutions. While the decontamination solutions passed through the vitrification melter, they did not have the typical impact of flushing contaminated equipment for decontamination purposes.

Because the vitrification melter molten glass pool remained around the 26-inch level until the time when the evacuated canisters were used to remove molten glass, the decontamination solutions in batch 76 and 77 gradually reduced the radionuclide concentrations in the glass pool as they were slowly fed into the vitrification melter and material from the glass pool was periodically poured into the HLW canisters. This effect can be seen in the falling dose rates for the last group of canisters produced as shown in Attachment 1.

Cs-137 Activity in the Vitrification Melter Before Use of the Evacuated Canisters

The information in Table 4-4 indicates that an estimated 8,668 Ci remained after processing the second batch of decontamination (flush) solution and prior to use of the evacuated canisters to empty the vitrification melter to the maximum extent technically and economically practical. The accuracy of this estimate can be assessed by use of alternative estimating methods.

Data recorded before evacuated canister operations began, including melter level data, show that approximately 2,230 kg of molten glass remained in the melter cavity after the final airlift and before use of the two evacuated canisters. The measured Cs-137 concentration in the glass shard samples taken from the canisters was 2.38E+03 μCi/g (WMG 2004b). An estimate of the Cs-137 activity in the 2,230 kg of molten glass present before use of the evacuated canisters – 2,230 kg multiplied by the 2.38E+03 μCi/g measured concentration – yields approximately 5,310 Ci. The total activity at 2.53E+03 μCi/g would be approximately 5,640 curies. This result shows that the estimate in Table 4-4 of the draft evaluation and Table 11 of the Perdue report on which it is based is conservative.

Attachment 3 describes another method that can be used to estimate the amount of Cs-137 remaining in the vitrification melter before use of the evacuated canisters. This method produced an estimate of 7,160 Ci, reinforcing the conclusion that the estimate in Table 4-4 and the Perdue report is conservative.

Changes to the Draft Evaluation

Changes are being made to the draft evaluation to reflect the results of evaluation of the NRC comment as described below.

NRC request (3): Describe if any measurements were taken between flushes to support assumptions regarding the efficiency of each individual flush.

DOE response: Please see the response to request (1). Available measurements include the concentrator feed makeup tank, the glass sample data, the canister dose rates, the evacuated canister weights, and the evacuated canister glass heights.

NRC request (4): Describe the volume of decontamination fluids used to flush material through the system and if this was enough to advance the residual sludge/molten glass remaining in the cavity.

DOE response: The volume of decontamination fluids amounted to approximately 240,000 gallons based on available records (Drake, et al. 2002), although this volume was reduced by more than 95 percent by evaporation[14].

Regarding whether this amount was enough to advance the residual sludge/molten glass remaining in the cavity, one has to consider the vitrification melter processes described previously. The flush solutions entered the vitrification melter slowly, like the HLW slurry. They were homogenized in the molten glass pool by convection currents associated with temperature differentials within the melter cavity. Glass was airlifted from this pool when the glass level rose about 0.7 inch above the level following the previous airlift. The airlifts continued until a canister was filled to approximately 90 percent of its capacity, then it was replaced by another empty canister. This process continued as batch 77 material entered the vitrification melter and the radionuclide concentrations in the molten glass pool gradually diminished as a result. During this process, it was also necessary to follow the appropriate glass-making recipe to ensure that the vitrified product would meet vitrified HLW waste form requirements (DOE 1996).

NRC request (5): Describe if any volume of glass was expected to be added to the cavity during the flushing process from the process lines, other tanks associated with the Vitrification Facility process.

DOE response: The decontamination solutions from all of the vitrification system flushes went to the concentrator feed makeup tank and ended up in the vitrification melter (WVNSCO 2002b).

NRC request (6): Describe if there is a portion of the cavity that cannot be readily discharged via flushing (e.g., stagnant portion on the bottom).

DOE response: As explained previously, the only way to remove material from the vitrification melter during operations was the airlift process, which was initiated when the glass pool level fell slightly below 26 inches, so material was not otherwise discharged via flushing. The evacuated canister system was designed to remove molten glass from the bottom part of the vitrification melter and this was done.

[14] The basis for the more than 95 percent estimate is comparison of the 240,000-gallon total flush volume with the 6000-gallon capacity of the concentration feed makeup tank, which was filled two times with decontamination (flush) solutions and glass formers.

Figure 2 shows the approximate position of the end of the snorkel with respect to the bottom of the cavity. As noted previously, drawings show that the lower end was positioned so it would be drawing molten glass from a point approximately six inches from the bottom of the cavity for the first evacuated canister and two inches from the bottom for the second one.

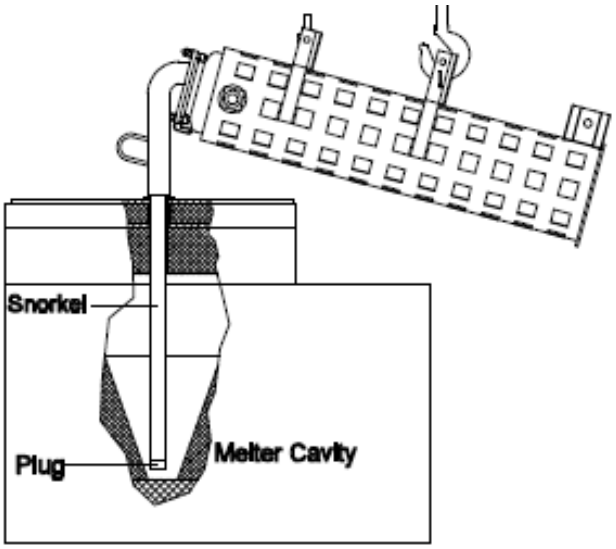

Figure 2. Evacuated Canister Snorkel Concept *(from May, et al. 2003)*

NRC request (7): Describe the uncertainty around the estimated 4,460 Ci remaining (e.g., from the uncertainty in the detection instrument used, or the uncertainty in the assumptions about remaining geometry).

DOE response: The uncertainty associated with the estimate from dose rate measurements is considered to be ±20 percent. The waste stream profile documentation (WVES 2010b) includes this uncertainty band in the table showing waste form concentrations in Bq/m^3.

As noted previously, the 4,570 Ci estimate produced by the dose-to-curie calculations (WMG 2004b) is conservative based on the results of an estimate made using the estimated 425 kg mass of residual material and the radionuclide concentrations in the glass samples.

Conclusions and changes to the draft evaluation: As noted previously, the primary conclusion from evaluation of this comment is that the estimates of flushing effectiveness given in Table 4-4 of the draft evaluation and Table 11 of the Perdue report on which Table 4-4 was based are conservative. Consideration of the details of the process also leads to the conclusion that it would be justifiable to consider each canister filled with flush solutions to represent a separate and distinct flush of the vitrification melter, although there are no plans to change the draft evaluation in this regard.

Section 4 of the draft evaluation is being revised to include additional information on the results of the alternative estimating methods described previously to support this conclusion. The discussion about additional flushing in Section 4.3 is also being revised to more clearly show why additional system flushing would not have been economically practical. Please see also the responses to comments MEP-5 and MEP-8 below.

Number: MEP-5

Subject: The draft WIR evaluation lacks a technical basis for the decision to use the evacuated canister after exactly three flushes.

Basis: NUREG-1854, Section 3.3.2, states that the reviewer should identify any removal goals DOE established before radionuclide removal began, and also to consider whether DOE considered modifications to the removal process to improve removal if termination is based on declining removal efficiency (NRC 2007). The efficiencies of the removal technologies are estimated, but specific removal goals for each technology are not indicated. It is not clear why DOE initially decided to use the evacuated canisters after the first three flushes, instead of implementing additional flushes prior to using the evacuated canisters. Also, it is not clear that a portion of glass was expected to remain which would not be easily evacuated using the canisters. In other words, it is not clear if the evacuated canisters performed to reasonable expectations/goals of the removal technology.

Path Forward: (1) Provide technical basis for why the evacuated canisters were used after exactly three flushes. One reference does not provide this basis (Perdue 2004); because the study was completed after the decision to use the evacuated canisters subsequent to three flushes had been implemented. (2) If the decision to use the evacuated canisters after only three flushes was based on declining removal efficiency, please specify if DOE considered modification of the flushing method to improve removal efficiency prior to using the evacuated canisters. (3) Also describe if there is a basis for the portion of material that was expected to remain at the bottom of the Melter that could not be readily evacuated using the canisters, and if this basis is consistent with the amount that remained. (4) If the evacuated canisters were expected to be capable of removing all glass, then discuss why additional evacuated canisters were not used.

NRC request (1): Provide technical basis for why the evacuated canisters were used after exactly three flushes. One reference does not provide this basis (Perdue 2004); because the study was completed after the decision to use the evacuated canisters subsequent to three flushes had been implemented.

> **DOE response:** Please see the response to comment MEP-4. The flushes were accomplished in accordance with the *HLW Processing Systems Flushing Operations Run Plan* (WVNSCO 2002b). This plan stated that “the primary objectives of the flushing are to transfer to the vitrification melter as much HLW as technically and economically practical and at the earliest opportunity so it could be vitrified.” It also identified expected post-flushing conditions for each component.
>
> The plan provided for flushing Tank 8D-1, Tank 8D-2, the waste header, Tank 8D-4, sludge mobilization system piping, portions of the supernatant treatment system, the low-level waste evaporator, the concentrator feed makeup tank, the melter feed hold tank, the submerged bed scrubber, and the vitrification melter. The flushing media included nitric acid, water, simulated waste, and glass formers. The concentrator feed makeup tank and the melter feed hold tank were required to be flushed by two passes using high-pressure spray.
>
> The plan described plans for the vitrification melter as follows:
>
> > “Pre-Flush Condition - The Melter is full of molten HLW glass. Since the Melter was started on non-radioactive glass, the radioactivity may not have penetrated very far into the refractory.
> >
> > Flushing Operations - After completion of HLW transfers from the Waste Tank Farm, feed with lower than normal radioactivity will be fed to the melter. The activity in these batches comes from normal recycle streams, such as, the LWTS Evaporator, flushing activities, canister

decontamination and the SBS [submerged bed scrubber]. The lower activity batches will serve to dilute the melter inventory of radioactive species. Although the recycle streams continue to be processed, and tank heels are adsorbed into the next batch, the final batch will consist primarily of simulated waste and glass formers.

Expected Condition After Flushes - It is expected that 3 to 6 inches of glass including a "noble metal sludge" will be left in the bottom *of* the Melter after the vacuum canisters.

Data Collection / Sampling - Sampling of glass shards from canisters are taken per SOP [standard operating procedure]. Videotaping *of* evacuated canister evolution is expected to occur."

The flushing plan acknowledged concerns over limited vitrification melter life but stated that "flushes may be repeated based on the results obtained." The melter feed hold tank was flushed three times instead of the required two times. A decision was then made that the flushing had been satisfactorily completed before it was concluded (WVNSCO 2002h and WVNSCO 2002i). Note that NRC staff had been consulted in connection with developing plans to complete the vitrification program, including the system flushes, as noted previously.

The Perdue study (Perdue 2004) was later commissioned to provide a retrospective evaluation as to whether key radionuclides had been removed to the maximum extent economically practical using the best available technology. This study concluded that further cleaning of the vitrification melter would not have been economically warranted based on the NRC's net social benefit criterion.

NRC request (2): If the decision to use the evacuated canisters after only three flushes was based on declining removal efficiency, please specify if DOE considered modification of the flushing method to improve removal efficiency prior to using the evacuated canisters.

DOE response: Consideration was given to processing another batch of decontamination solution. This batch would have been generated by an acid wash of the vitrification pit floor intended to move radioactive contamination into the north sump of the pit and subsequently to the concentrator feed makeup tank for additional melter feed and source term reduction. This approach was not used because of concerns over the acid "attacking the integrity or functions of the support structures, as well as concerns over processing the volume of acid used." (Kocialski 2003)

NRC request (3): Also describe if there is a basis for the portion of material that was expected to remain at the bottom of the Melter that could not be readily evacuated using the canisters, and if this basis is consistent with the amount that remained.

DOE response: As noted previously, the *HLW Processing Systems Flushing Operations Run Plan* (WVNSCO 2002b) indicated that three to six inches of glass were expected to be left in the bottom of the melter cavity, the snorkel intake with the second evacuated canister was positioned about two inches from the cavity bottom, and it was determined that approximately eight inches of glass remained. The evacuated canister system performed as planned (May, et al. 2003) but did not empty the melter cavity quite to the extent expected.

NRC request (4): If the evacuated canisters were expected to be capable of removing all glass, then discuss why additional evacuated canisters were not used.

DOE response: As indicated, the evacuated canister system was expected to leave three to six inches of glass at the bottom of the vitrification melter cavity and left slightly more than that.

Preparations had been made to use two evacuated canisters because two were expected to be more than sufficient to remove all of the molten glass in the melter cavity within the design limitations of the evacuated canister system[15]. It is unlikely that a third evacuated canister would have been useful since the snorkel was positioned when filling the second canister to reach within two inches of the cavity bottom.

Conclusions and changes to the draft evaluation: It would be useful for clarity to discuss the basis for accomplishing and concluding the flushing. Changes will be made to the draft evaluation to address these matters.

[15] The volume of glass remaining in the vitrification melter cavity was expected to fill one canister to the approximately 80 percent level and the second canister was made available as a contingency (Petkus 2002a)

Number: MEP-6

Subject: The cost benefit analysis used in the evaluation does not sufficiently discuss potential benefits associated with averted long-term dose to members of the public, including inadvertent intruders.

Basis: NUREG-1854, Section 3.3.2, states that the reviewer should identify any removal goals DOE established before radionuclide removal began, and also to consider whether DOE considered modifications to the removal process to improve removal if termination is based on declining removal efficiency (NRC 2007). The efficiencies of the removal technologies are estimated, but specific removal goals for each technology are not indicated. It is not clear why DOE initially decided to use the evacuated canisters after the first three flushes, instead of implementing additional flushes prior to using the evacuated canisters. Also, it is not clear that a portion of glass was expected to remain which would not be easily evacuated using the canisters. In other words, it is not clear if the evacuated canisters performed to reasonable expectations/goals of the removal technology.

Path Forward: Provide a discussion of the potential benefits in terms of reduction in dose to a member of the public, including to a potential inadvertent intruder from additional removal technologies. If DOE considers the reduction in doses resulting from additional removal to be negligible, DOE should provide a discussion of why they are negligible, which may include a comparison to the costs of removal.

DOE response: The potential benefits of additional key radionuclide removal in terms of a reduction in dose to a member of the public would be insignificant (1) during storage at the WVDP, (2) during transportation to the LLW disposal facility, (3) during disposal at the LLW disposal facility, and (4) to a potential inadvertent intruder in the area of the closed disposal facility. Any minimal benefits to members of the public from additional decontamination would have been far outweighed by the substantial worker radiation dose and monetary costs required to perform this additional decontamination.

Onsite Storage, Transportation, and Disposal

Risks to members of the public from the vitrification melter waste package during temporary storage at the WVDP were analyzed in the *West Valley Demonstration Project Waste Management Environmental Impact Statement Supplement Analysis* (DOE 2006). This analysis showed that an accident involving dropping the melter waste package onsite would produce only 0.049 mrem to the maximally exposed offsite individual, a small fraction of the 10 mrem per year limit.

Risks to members of the public from transportation to the LLW disposal facility were also analyzed in the Supplement Analysis. The risks to members of the public from radiation during rail shipment to the Nevada National Security Site was expressed as 0.0000068 latent cancer fatalities. The risk for shipment to the Waste Control Specialists facility in Texas was lower because of the shorter distance to that facility.

Risks to member of the public during disposal of the waste package are expected to be very low as discussed on page 57 of the draft evaluation. One reason for the low risks during storage, transportation, and disposal is the low gamma radiation levels on the outside surfaces of the vitrification melter waste package. As noted on page 56 of the draft evaluation, these levels range from 2 to 6 mrem per hour in 2004 and will be lower still after the waste package is grouted prior to shipment offsite.

Further decontamination of the vitrification melter would have had minimal impacts on these risks and this decontamination effort would have resulted in significant worker radiation exposure. It was

estimated that one additional month of vitrification melter operation would have resulted in worker exposure of 7.36 rem (Perdue 2004, Table 2). As discussed in Attachment 4, the monetary costs of processing an additional batch of decontamination solutions in the vitrification melter to produce a single additional canister would have been in the order of $2.5 million.

Inadvertent Intrusion into the Closed LLW Disposal Site

Section 5.2.3 of the draft evaluation addresses protection of individuals from inadvertent intrusion. This section shows that disposal of the vitrification melter waste package at Nevada National Security Site's Area 5 LLW disposal facility would have a negligible impact on the dose to an advertent intruder at that facility. This conclusion is based on the results of a special performance assessment that evaluated disposal of the vitrification melter waste package. Given this result, there would be negligible, if any, benefits in terms of reduced dose to an inadvertent intruder from further reduction of the radioactivity in the vitrification melter waste package.

A similar special performance assessment was not performed for the Waste Control Specialists (WCS) LLW disposal facility in Texas because that facility was not in operation at the time the draft evaluation was prepared (and is still not yet in operation). Section 5.2 shows that the estimated doses to an inadvertent intruder at that facility would be low. The WCS facility provides for a cover depth over the waste in excess of five meters (WCS 2007). Given this design and the results of the special performance assessment for disposal of the vitrification melter at the Nevada National Security Site, it is reasonable to expect that disposal of the melter waste package at the WCS facility would have a negligible impact on dose to an inadvertent intruder at that facility. Therefore there would be negligible, if any, benefits associated with further reduction of the radioactivity in the vitrification melter waste package if it were to be disposed of at the WCS facility.

Because there would be negligible, if any, benefits from use of additional removal technologies, no cost-benefit study related to this matter is necessary.

Conclusions and changes to the draft evaluation: The risks associated with onsite storage, transportation, and disposal of the vitrification waste package without further decontamination are very small, as is the risk to a hypothetical intruder in the areas of the closed disposal facility. The costs of further decontamination would far outweigh the small potential benefits as discussed in Attachment 4.

No changes to the draft evaluation in connection with this matter are planned, other than the changes described in Attachment 4.

Number: MEP-7

Subject: The basis for applying fixative coating to the outside of the Melter versus removal of key radionuclides to the maximum extent practical is not sufficient.

Basis: The draft WIR evaluation describes the use of a fixative coating on page 24, in the "Data Used in Characterization" section, last sentence, "Both types of contamination are now considered to be fixed because Polymeric Barrier System, a latex fixative coating, was applied to the outside of the Vitrification Melter before it was placed in the shipping container." A fixative coating helps to prevent spread of contamination, but it does not remove contamination.

Path forward: (1) DOE should explain how fixing contamination on the outside of the Melter coincides with removal to the MEP. (2) DOE should summarize any efforts that were made to decontaminate the Melter prior to fixing the contamination with the coating.

NRC comment (1): DOE should explain how fixing contamination on the outside of the melter coincides with removal to the MEP.

DOE response: The accessible external surfaces of the vitrification melter were decontaminated by flushing with water as explained in response to the second part of this comment. The use of the fixatives in lieu of additional decontamination coincides with removal of key radionuclides to the maximum extent practical because any efforts to further decontaminate the outside of the vitrification melter would have been inconsistent with as low as reasonably achievable (ALARA) requirements for the following reasons:

- The residual radioactivity on the outside of the vitrification melter has been determined to be only 0.11 percent of the total radioactivity in the waste package (WMG 2004b), which is an insignificant amount.
- The vitrification melter waste package has been accepted for disposal at the Nevada National Security Site in its present condition, subject to stabilization of the melter and the waste package with grout (DOE 2010a).
- Additional decontamination of the outside of the vitrification melter before it was placed in its waste package would not have been cost-effective because the melter can be disposed of as LLW without such decontamination.
- Additional decontamination of the outside of the vitrification melter in its present packaged condition would be expensive, involve unnecessary worker radiation exposure, and produce no benefits[16].

NRC comment (2): DOE should summarize any efforts that were made to decontaminate the melter prior to fixing the contamination with the coating.

[16] There are many drawbacks to additional decontamination of the exterior surfaces of the vitrification melter. First, it would have to be removed from its shielded container. This effort would entail moving the shielded container to a suitable work area or installing a containment enclosure over it to support accomplishing this work outside. Moving the container is a considerable undertaking because of its great weight. Decontaminating the melter exterior surfaces remotely inside an area such as the vitrification pit would be time consuming and expensive. Most suitable methods would produce a secondary waste stream. Packaging this waste for disposal and disposing of it would entail additional cost and worker radiation exposure. And even if additional decontamination were to be performed, it would produce negligible, if any, benefits for the reasons stated in the text.

DOE response: The *HLW Processing Systems Flushing Operations Run Plan* (WVNSCO 2002b) provided for flushing with pressurized water the accessible areas of the vitrification pit, including the outside surfaces of the vitrification melter and other equipment. The flush water and the removed contamination were collected and transferred to the concentrator feed makeup tank so they could be vitrified with other decontamination solutions. Efforts were made to determine the effectiveness of the flushing by visual inspection but the resultant effect was difficult to access because staining of the base metal masked the capability of the closed-circuit television cameras to determine cleanliness (Lachapelle 2003).

Conclusions and changes to the draft evaluation: It would be useful for clarity to mention in the draft evaluation the efforts made to decontaminate the exterior of the vitrification melter and this will be done.

Number: MEP-8

Subject: The equation for Melter activity removed by the ith flush does not appear to incorporate data from the first three flushes.

Basis: NUREG-1854, Section 3.4, states that the reviewer should verify that the extent of radionuclide removal that could be achieved is not underestimated. The additional activity removed by the ith flush are estimated using an equation that assumes a log-reciprocal relationship between the jth month of Vitrification Facility operation and the reduction factor (Perdue 2004). With each additional flush the amount of activity removed per unit time will decrease. It is reasonable to expect diminishing returns with additional application of additional flushes, but there exists a difference between the removal efficiency of the first three flushes and the next hypothetical twelve flushes. DOE does not describe a change in the flushing process for the hypothetical flushes, so one would not expect such a stark contrast in the removal efficiency.

Path Forward: (1) Explain if the analysis to predict removal efficiency of additional flushes applied knowledge gained from the actual removal efficiencies of the first three flushes, and if it did not why it did not. (2) Provide the amount of time it took to complete each of the first three flushes, as well as cost information for the first three flushes to the extent that it is available. (3) Explain the difference in the removal efficiency of the first three flushes (determined to have removed 46% of the remaining activity), with that of the fourth flush, which was only predicted to remove 0.5% of the remaining activity, or 22 Ci. (e.g., explain if there was a difference in the assumptions regarding the flushing process or the amount of glass formers that would be used with each flush).

> **DOE response:** Consideration of this comment and evaluation of other NRC comments related to key radionuclide removal have led to the conclusion that changes to the draft evaluation are needed regarding the effectiveness of the hypothetical additional flushes.

Conclusions and changes to the draft evaluation: Section 4.3 of the draft evaluation is being revised accordingly. Attachment 4 describes the planned changes to this section.

Number: WC-1

Subject: The explanation of how averaging over void volumes was applied for purposes of defining waste classification is not clear.

Basis: Page 8 of the draft WIR evaluation, 1st paragraph states, "This grout will not increase the waste disposal volume and was not considered in the classification of the waste." The statement is misleading with respect to the classification of the waste, because internal void volume within the Melter that will be filled with grout was considered in the classification of waste. Page 65 of the draft WIR states that the weight used in the calculation was 106,000 lbs and the volume used was 750 ft3 (21 m3). A reference describes how this volume is derived from the outer dimensions, which would include any void spaces on the inside of the Melter (WSMS, 2008).

Path Forward: (1) DOE should discuss whether large void volumes remain in the Vitrification Melter that must be filled with grout, and the void spaces that are expected to remain after the Melter is grouted. (2) If there are large void spaces, DOE should indicate why it is acceptable to average over these large void volumes or justify why the averaging volume is not important to the sum of fractions (e.g., based on its sensitivity analysis or the fact that transuranics dominate the waste classification calculations since transuranics are based on concentration limits that average over the mass, not the volume of waste).

NRC comment (1): DOE should discuss whether large void volumes remain in the vitrification melter that must be filled with grout.

> **DOE response:** As noted in the draft evaluation, voids in the vitrification melter will be filled with grout prior to shipment offsite. The total volume of the melter is approximately 750 cubic as noted in the draft evaluation. Voids in the vitrification melter comprise approximately 378 cubic feet.

NRC comment (2): If there are large void spaces, DOE should indicate why it is acceptable to average over these large void volumes or justify why the averaging volume is not important to the sum of fractions (e.g., based on its sensitivity analysis or the fact that transuranics dominate the waste classification calculations since transuranics are based on concentration limits that average over the mass, not the volume of waste).

> **DOE response:** The presence of the voids is unimportant to the Class C sum-of-fractions estimate for the following reasons:
>
> - The classification is controlled by transuranic radionuclide concentrations in nCi/g that do not depend on vitrification melter volume.
> - As indicated in note (3) to Table 6-1 of the draft evaluation, the sensitivity study, which included a volume reduction by 50 percent to 375 cubic feet, showed that transuranic radionuclides would still dominate the waste classification calculation results by a wide margin. The 375 cubic foot value is essentially the same as the actual 378 cubic foot void volume.

Conclusions and changes to the draft evaluation: It would be useful for clarity to discuss the presence of void spaces in the vitrification melter and how they do not impact waste classification. Information to this effect is being included in the evaluation.

Number: WC-2

Subject: The bases for the list of radionuclides considered in the waste classification are not clear.

Basis: Table 1 of 10 CFR 61.55 contains concentration limits for long-lived radionuclides including a specific class of radionuclides, alpha-emitting transuranic radionuclides with half-lives greater than 5 years. The following transuranic radionuclides with half-lives greater than five years are listed in the 1987 inventory of the sludge and supernatant but are neither carried forward to the inventory of the residual glass in Table 2-2, the inventory in the Melter characterization reference (WMG 2004b), nor are they included as part of the waste classification in Table 6-1: Pu-242, Am-242m, Cm-245, and Cm-246.

Path Forward: DOE should address why the transuranics with half-lives greater than 5 years that were listed in the sludge/supernatant inventory, were not considered in the waste classification.

DOE response: Table 7 shows estimated concentrations of these radionuclides in the vitrification melter, their impact on the Table 1 sum of fractions, and the basis for these estimates.

Table 7. Additional Radionuclides

Radionuclide	nCi/g	Table 1 Fraction	Basis
Pu-242	4.67E-04	4.67E-06	Note (1).
Am-242m	1.90E-03	1.90E-05	Note (2).
Cm-245	2.25E-01	2.25E-03	Note (1).
Cm-246	3.66E-02	3.66E-04	Note (1).

NOTES: (1) Concentration from WVES 2010b, Table 4-1 of Technical Basis Document.

(2) Using the ratio of Am-242m to Am-241 from Rykken 1986, Table 22, and the measured Am-241 concentration from WVES 2010b, conservatively assuming that alpha radiation represents 0.5 percent of emissions.

These radionuclides were not included in Table 6-1 due to their low activity concentrations, which would have a negligible impact on the Table 2 sum of fractions used in classification of the waste package. The Table 1 sum of fractions changes from 0.949 to 0.951 and the Table 2 sum of fractions remains the same at 0.0459.

Conclusions and changes to the draft evaluation: It would be appropriate for transparency to include these radionuclides in Table 6-1 and the table is being revised accordingly.

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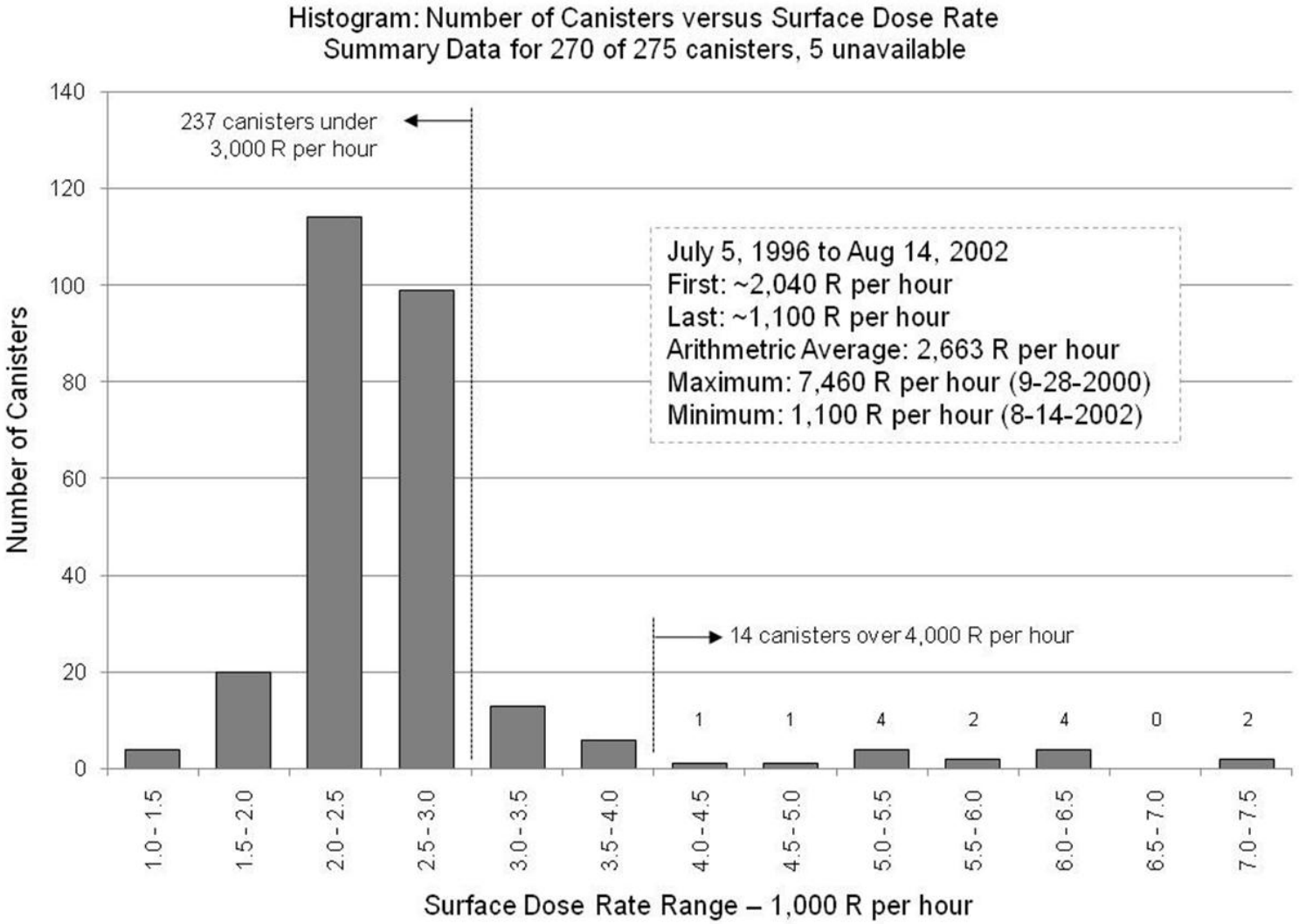
ATTACHMENT 1 – GRAPH OF HLW CANISTER DOSE RATES

The following histogram shows gamma dose rates measured on the filled HLW canisters based on data from vitrification records (WVES 2011).



ATTACHMENT 2 – CALCULATIONS OF VITRIFICATION MELTER GLASS HEEL

1.0 INTRODUCTION

This calculation was performed to provide an independent assessment of the depth of the remaining glass in the cavity of the vitrification melter using available data.

2.0 CALCULATION

Based on the measured height of glass inside each:

- Evacuate Canister 1 (WV-997) contains about 27,526 cubic inches of glass
- Evacuated Canister 2 (WV-998) contains about 20,227 cubic inches of glass

Thus, this dimensional data taken for each canister indicates the evacuated canisters removed about 47,753 cubic inches of glass from the melter cavity.

Two load cells weighed the evacuated canisters before and after use. Assume the load cells were reading pounds (units are not noted on the data sheet, WVNSCO 2002c).

Empty, WV-997 weighed 2,700 pounds, filled, 5,410 pounds indicating it contained 2,710 pounds of glass (1,229.02 kg).

Empty, WV-998 weighed 2,590 pounds, filled, 4,680 pounds indicating it contained 2,090 pounds of glass (947.85 kg).

Thus, this data indicates that the evacuated canisters removed about 2,177 kg of glass. Assuming 2.4 g/cc, this equates to 55,350 cubic inches of glass removed. Assuming 2.6 g/cc, this equates to 51,093 cubic inches of glass removed. (The Mass Balance sheets included with each canister record used 2.4 kg/L as the glass density (WVNSCO 2002c); WMG used 2.6 g/cc as indicated in WMG 2004b.)

Physical dimensional measurements (WVNSCO 2002a and WVNSCO 2002b) indicate 47,753 cubic inches of glass removed. Nominally these all indicate the same basic number with the 47.7K being the least volume removed.

It seems L. Petkus assumed the melter cavity was at 26 inches at the start of the evacuation operation (Petkus 2002b).

The data sheet for the last canister made (WV-412 dated 8-14-2002) indicates that at the end of making canister 412, the level in the melter was 24.38 inches according to instrument LIX-2009/2009a. It is highly likely there was no more feed from the MFHT after 8-14-2002. (It was the end of operations and both the CFMT and the MFHT were as empty as they could physically be.) The Evacuated Canister data package does not document the starting melter level. There is probably some measuring uncertainty associated with LIX-2009/2009a.

Physically, each snorkel assembly had a built-in hard stop that located the end of the snorkel nominally six inches from the bottom of the melter cavity for the first canister and four inches from the bottom of the cavity for the second canister (drawings PNL-673-02 and PNL-673-05, WVNSCO 2002b and WVNSCO 2002a. respectively) Thus if 100% effective, the snorkel would have left a two inch heel.

The active melter cavity is comprised of two basic geometric shapes. The first 16 inches is a rectangular truncated pyramid (that is also inverted with the small side down). From the 14 by 25 inch rectangular base of this pyramid (the bottom of the melter cavity), the north and south walls

slope upward from the 14 inch sides steeply at 65.15 degrees and the east west walls slope upward from the 25 inch side of the base at a shallower 30.65 degrees. At 16 inches above the base, the 14 by 25 rectangle has expanded to a 68 inch by 38.82 inch opening respectively. On top of this truncated inverted rectangular pyramid is a trapezoidal prism with a fixed width in the east west direction (68 inches) and an expanding width in the north south direction continuing to follow the steep upward slope of 65.15 degrees from the pyramid below.

When filled to a 26 inch depth, the theoretical internal volume associated with this shape is comprised of the full 16 inches of the rectangular truncated pyramid and 10 inches of the trapezoidal prism. The theoretical volume of this full cavity (26 inches measured from the cavity bottom) is calculated to be 52,157 in^3.

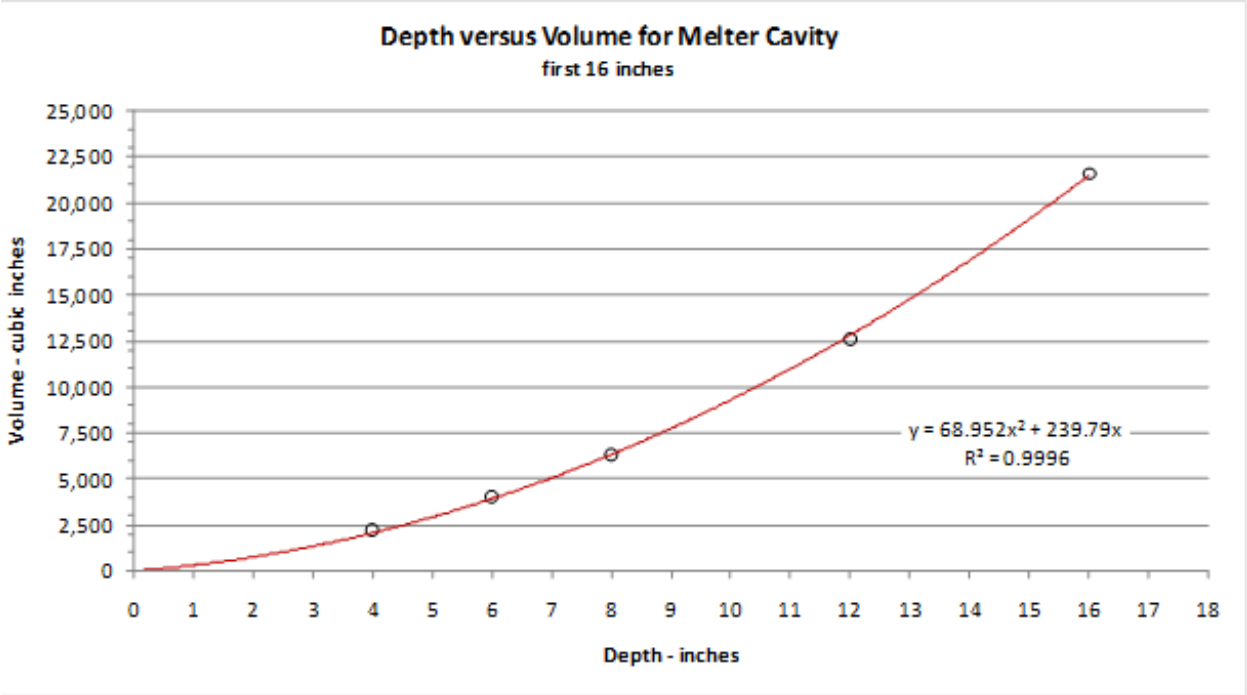


Figure 2-1. Depth vs. Volume for Melter Cavity

Section 2 page 10 of the Vitrification System Description document (Petkus 2002a) notes *"The nominal operating volume of glass in the melter is 860 liters."* This converts to 52,480 in^3.

The first evacuation canister removed 27,500 in^3 of glass (based on physical dimensional data, the lowest and thus most conservative number) and the second removed 20,200 in^3 for a total removed volume of at least 47,700 in^3. This indicates that after the evacuation canisters there was nominally 4,500 in^3 of glass (52,200 - 47,700 = 4,500) remaining as a heel in the melter.

Given the geometry of the melter cavity this equates to a heel about 6.5 inches deep. For information, an 8 inch heel is estimated to be 6,330 in^3, 41% more than the calculated value.

3.0 REFERENCES

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ATTACHMENT 2 – CALCULATIONS OF VITRIFICATION MELTER GLASS HEEL

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ATTACHMENT 3 – ALTERNATIVE ESTIMATE OF CS-137 PRESENT BEFORE USE OF EVACUATED CANISTERS

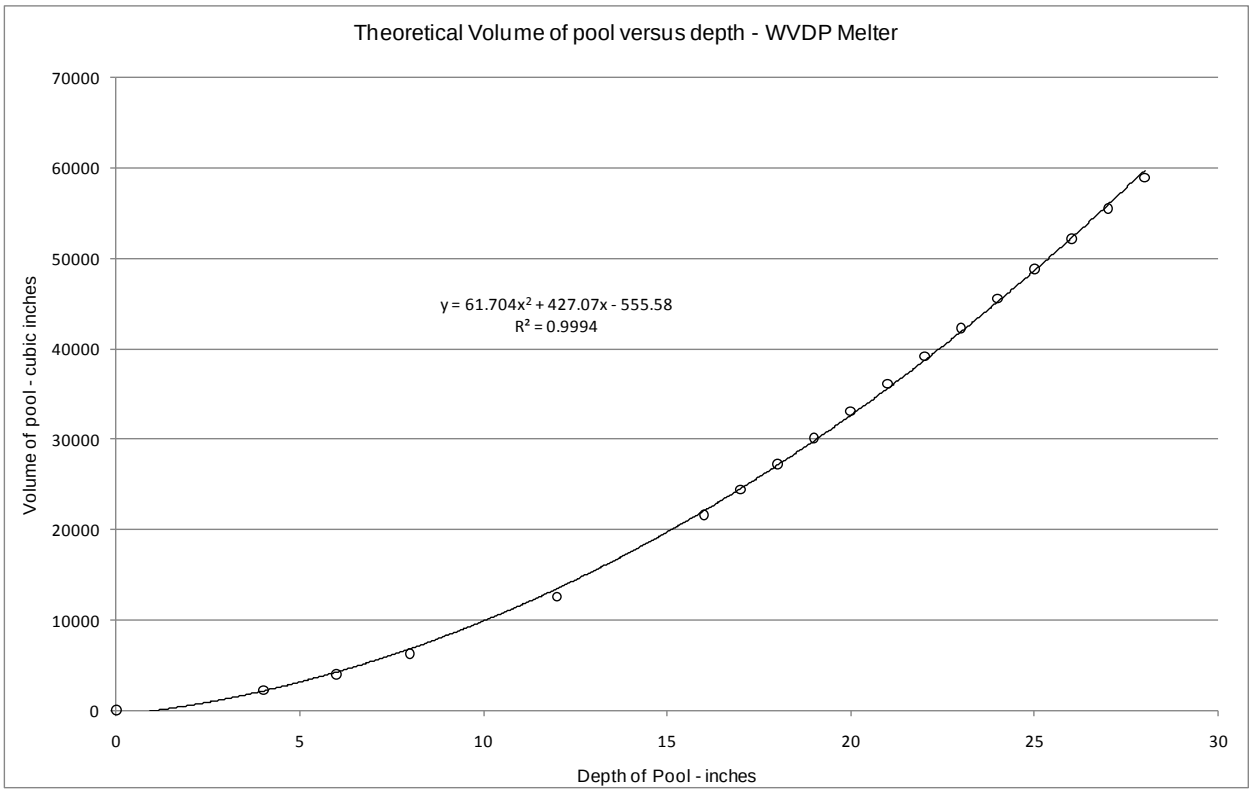
1.0 INTRODUCTION

The following calculation addresses the observed decrease in radioactivity in the glass and the residual glass heel of the West Valley melter.

2.0 CALCULATION

Batch 75 was the last "pure" HLW batch sent to the melter. Batches 76 and 77 included the collected residuals from flushing the entire HLW system. Batch 76 was less contaminated than Batch 75 and more contaminated than Batch 77.

The melter operated with a nominal glass level of 26 inches from the bottom of the melter cavity. At this level, the theoretical volume is 856.4 liters. With a density of 2.4 grams per cm^3, this would represent 2,055 kg of slowly churning molten glass. The cavity volume in cubic inches versus inch of depth is shown in the graph below. For conversion, there are 16.4 cm^3 in a cubic inch.



The last canister containing Batch 75 material was canister WV-405. When completed, this canister was estimated to contain 2,010 kg of glass and 32,000 curies. This indicates a nominal curie density of 16.0 curies per kilogram of glass at this time in the process. At the end of processing canister WV-405, the melter was at 24.7 inches. This is estimated to be 798 liters and if filled with glass at this curie density would contain 29,860 curies. Thus the starting point for the melter processing flush solutions (i.e., the melter flushing evolution) was about 30,000 curies.

ATTACHMENT 3 – ALTERNATIVE ESTIMATE OF CS-137 PRESENT BEFORE USE OF EVACUATED CANISTERS

Batches 75, 76 and 77 were each sampled in the Concentrator Feed Make-up Tank after preparations were completed prior to transferring the material to the Melter Feed Hold Tank and then the Melter to make glass filled canisters. A comparison of this data indicates that Batch 76 contained nominally 51.5% of the radioactivity of Batch 75. This reduction is dominated by a reduction in the amount of Cs-137. This sample data showing the reduction in radioactivity of these batches, based on those common radioisotopes measured for each batch is shown below. This data would indicate that as glass, Batch 76 material would be expected to have about 51.5% of the radioactivity of Batch 75 material. This would represent a nominal curie density of 8.24 curies per kilogram of glass. Thus if the melter cavity were filled to a 26 inch level solely with this material, 15,380 curies would be present.

Common Isotopes	Batch 75	Batch 76	Batch 77
Am-241	3.82E+00	2.08E+00	4.80E-01
Co-60	2.96E-01	2.71E-01	2.46E-02
Cs-137	1.16E+04	6.25E+03	1.39E+03
Pu-238	1.27E+00	5.79E-01	1.04E-01
Pu-239+240	5.36E-01	2.10E-01	4.03E-02
Sr-90	8.70E+02	1.90E+02	4.27E+01
Total Activity Conc.	12,520.0	6,446.5	1,430.0
Relative Factor	1.000	0.515	0.114

With some allowance for system “inertia”, a general confirmation of these decreasing properties as well as evidence of batch to batch mixing can be seen in a comparison of the surface dose rate reductions from the final progression of canisters produced by the WVDP Melter.

Canister No	Batch	Surface Dose	Comparison
WV-405 (#268)	Last with Batch 75	2,800 R/hr	1.00 ref
WV-409 (#272)	Last with Batch 76	2,060 R/hr	0.74
WV-412 (#275)	Batch 77	1,100 R/hr	0.39
WV-498 (E.C.#2)	Residual	500 R/hr	0.18

Canister WV-410 was the first canister made with 100% batch 77 material. It was also one of the randomly selected canisters for which shard analysis was performed. (Twenty eight canisters were randomly selected for shard radiochemistry analysis to explicitly measure Cs-137 and Sr-90 values for use in package characterizations.) Canister WV-410 is estimated to contain 2,024 kg of glass and 7,910 curies (WVNSCO 2002a). This indicates a curie density of the glass at this time in the process of 3.910 curies per kg of glass.

ATTACHMENT 3 – ALTERNATIVE ESTIMATE OF CS-137 PRESENT BEFORE USE OF EVACUATED CANISTERS

It is also known that the Evacuation Canisters contain glass with a curie density of nominally 2.5 curies per kg of melter glass (from data in WVNSCO 2002a and WVES 2011). When combined with the other data discussed above, this indicates the decrease in radioactivity associated with the melter glass as shown in the table below.

Basis	**Curie Density of Melter Glass**
Batch 75 (Canister WV-405)	16.0 curies per kg of glass
Batch 76 (CFMT Comparison data)	8.2 curies per kg of glass
Batch 77 (Canister WV-410)	3.9 curies per kg of glass
Evacuation Canisters	2.5 curies per kg of glass

Canister WV-412 was the last production canister made (i.e., canister no. 275) at West Valley. When it was filled, the melter level was at 24.38 inches (WVNSCO 2002b). With a curie density of 3.91 curies per kg of glass, and a glass density of 2.4 gms per cm^3 it is estimated that the just prior to the deployment of the evacuation canisters the melter held 1,830 kg of glass containing nominally 7,160 curies in 1,830 kg of molten glass.

The most limiting removal data (i.e., predicting the least amount of glass removal from the melter cavity) for the evacuation canisters is based on the physical dimensional data associated with the nominal fill height of the glass within the canisters. (The data based on canister before and after weights predicts more glass was removed.) Evacuation canister 1 is predicted to remove about 27,500 in3 of glass. With a density of 2.4 gm per cm3 this equates to 1,080 kg. Evacuation canister 2 is estimated to have removed 20,225 in3 or 795 kg. Each of these canisters was sampled. The sampling indicates Evacuation Canister 1 has a curie density of 2.45 ci per kg of glass and Evacuation Canister 2, 2.61 Ci per kg (from data in WVNSCO 2002a and WVES 2011). This indicates that the Evacuation Canisters conservatively removed at least 4,700 curies from the cavity containing nominally 7,160 curies. These data indicate that the residual activity in the melter heel is 2,460 curies.

Point in Sequence	Curies	Remarks
Starting point (End of Batch 75)	30,000	
End of Batch 76	15,380	Some allowance for the presence of higher activity material from prior batch is reasonable.
End of Batch 77(Before Evacuation Canisters)	7,160	
After Evacuation Canisters	2,460	

The methodology used by WMG to characterize the curies in the heel is based on the use of the dose rate data and a conservative model to estimate that over 4,000 curies remain in the heel in nominally 300 kg of residual glass. This equates to 13.3 curies per kg of melter glass, an obviously high (conservative) number based on the above data.

ATTACHMENT 3 – ALTERNATIVE ESTIMATE OF CS-137 PRESENT BEFORE USE OF EVACUATED CANISTERS

3.0 REFERENCES

WVNSCO 2002a, *Radionuclide Inventory for HLW Canistered Waste Form Production Records*, WVNS-CAL-325, Rev 2. West Valley Nuclear Services Company, West Valley, New York, February 25, 2002.

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WVES 2011, *Data package with Microsoft Excel spreadsheets of HLW canister dose rates and analytical data from samples collected from the Concentrator Feed Makeup Tank for Batches 10 – 77.* West Valley Environmental Services, West Valley, New York, April 26, 2011.

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The following changes are planned to Section 4 of the draft evaluation in regard to flushing of the vitrification melter. The reference designations will be finalized as the waste determination basis document is prepared. Changes from the March 2011 version of the draft evaluation appear in red.

4.2.2 Methods Considered

The design and operation of the Vitrification Melter posed unique challenges from a decontamination standpoint. If the melter were to be shut down without first being decontaminated, the molten glass containing HLW would immediately harden. The resulting condition would amount to a steel box containing a substantial amount of key radionuclides incorporated into hardened glass that would be exceedingly difficult to remove.

With this in mind, DOE considered a variety of methods to remove key radionuclides from the Vitrification Melter as discussed below. The best methods would involve reducing key radionuclide concentrations in the melter molten glass pool and then extracting as much of the molten glass as practical before melter shutdown.

Methods considered included those discussed in DOE's Decommissioning Handbook (DOE 1994). This handbook describes a wide range of technologies used for decontamination at DOE sites. Some of these methods – such as vacuuming, flushing with water, grinding, grit blasting, and milling – are widely used industrial technologies. Others are innovative technologies developed with the support of the DOE Office of Science as part of a continuing program to improve methods used in decontamination and decommissioning work. Some innovative decontamination technologies have been tested in field demonstration projects sponsored by DOE as described in innovative technology summary reports issued by the DOE Office of Science and Technology, and these were considered. However, none of these technologies was identified as specifically applicable to a vitrification melter.

DOE's Decommissioning Handbook identifies advantages and disadvantages of decontamination technologies in various applications. The most relevant application for decontaminating the Vitrification Melter internals was "Embedded Material and Some Oxide Surfaces." The DOE Handbook considered five technologies to be highly effective in this application: (1) hydroblasting, (2) ultra-high pressure water grit blasting, (3) drill and spall, (4) paving breaker/chipping hammer, and (5) expansive grout.

DOE developed a vacuum extraction system using evacuated canisters to remove the majority of residual dilute molten glass from the Vitrification Melter prior to shutdown. This system could only be utilized before the contents of the Vitrification Melter were allowed to solidify after the heaters were secured and the Melter allowed to cool.

After considering available options, the WVDP determined that three fundamental approaches existed for potential removal of additional key radionuclides from the Vitrification Melter: (1) processing solutions with lower radionuclide concentrations, (2) mechanical means (including the five technologies the DOE Handbook identified as high effective in the application just discussed) , and (3) dismantlement.

The methods evaluated in detail were:

- Processing vitrification system decontamination solutions;
- Mechanical – evacuated canisters, ball milling; and
- Dismantlement – size reduction and waste segregation.

The objective of each of these potential methods was to remove residual material in the equipment, including key radionuclides, to the extent that was technically and economically practical. The remaining available options were considered to be experimental[17] on such an application as a vitrification melter. In 2008, DOE reconsidered use of a dismantlement approach to determine whether using this approach with present-day technologies would be technically practical as discussed in Section 4.2.6.

4.2.3 Processing Vitrification System Decontamination Solutions

A technically and economically practical method for removing key radionuclides from the Vitrification Melter involved processing decontamination solutions from other vessels. This method was used successfully while the Vitrification Facility was still operational[18].

Vitrification System Flushing

In late 2000, DOE commissioned a Vitrification Completion Team composed of representatives from DOE, NYSERDA, West Valley Nuclear Services, and NRC to review issues surrounding the ability to complete vitrification operations (VCT 2001). This team developed an approach to retrieving waste from the underground waste tanks, washing and characterizing the residual tank materials, and flushing the vitrification system prior to completing a controlled shutdown of the melter.

The vitrification system flushes involved flushing Tank 8D-1, Tank 8D-2, the waste header, Tank 8D-4, Sludge Mobilization System piping, portions of the Supernatant Treatment System, the Low-Level Waste Evaporator, the Concentrator Feed Makeup Tank, the Melter Feed Hold Tank, the Submerged Bed Scrubber, and the Vitrification Melter. The Concentrator Feed Makeup Tank and the Melter Feed Hold Tank were required to be flushed by two passes using high-pressure spray. For the Vitrification Melter, this process involved feeding material with lower than normal radioactivity concentrations to reduce the melter radionuclide inventory. (WVNSCO 2002a)

The flushing plan (WVNSCO 2002a) acknowledged concerns over limited Vitrification Melter life but stated that "flushes may be repeated based on the results obtained." Based on this criterion, the Melter Feed Hold Tank was flushed three times instead of the required two times. A deliberate decision was made that the flushing had been satisfactorily completed before the evacuated canisters were used and the Vitrification Melter shutdown (WVNSCO 2002c and WVNSCO 2002d).

A detailed description of the vitrification system flushing accomplishments appears in the *Report on Deployment of Miscellaneous Tanks and Piping Cleaning Equipment and Methodology* (WVNSCO 2002b). This report includes photographs that demonstrate the effectiveness of the flushing on the internals of the Concentrator Feed Makeup Tank and the Melter Feed Hold Tank. Additional flushing of these vessels and other vitrification system equipment was determined to be

[17] DOE considered it necessary to use a proven, mature technology to avoid unnecessary development costs and the inherent uncertainties that would have been associated with an experimental technology.

[18] Nitric acid and demineralized water were used in the flushes.

unnecessary based on the visual inspections that showed the vessel interiors to be clean and free of visible waste, and based on dose rate and radionuclide concentration data (WVNSCO 2002c).

Vitrification Melter

It was not feasible to literally flush the Vitrification Melter like the vessels used to feed slurry to the melter because the melter had to remain in operation to avoid hardening of the molten glass in the melter cavity. A high-pressure water spray apparatus could not be used inside the melter while it was in operation, as had been done with the Concentrator Feed Makeup Tank and Melter Feed Hold Tank.

Rather, decontamination solutions used to flush these two vessels were slowly fed to the melter like HLW slurry and thereby significantly reduced the radionuclide concentrations in the melter. This concentration reduction substantially reduced the amount of residual radioactivity in the glass that could not be removed from the vitrification melter by the second decontamination method used – the evacuated canister system, which is discussed below.

The effectiveness of this method in decontaminating the melter – that is, in reducing the concentrations of key radionuclides in the molten glass pool before use of the evacuated canisters – can be determined by comparing canister dose rates. These dose rates are proportional to the Cs-137 concentration in the material inside the canister[19]. Figure 4-1 shows how canister dose rates dropped after the decontamination flush solutions entered the melter glass pool.

As can be seen in Figure 4-1, canister dose rates started falling while batch 76, the first of the two batches of decontamination solutions, was being fed to the vitrification melter. They dropped further as batch 77 was being fed to the melter.

The decontamination factor associated with this process can be estimated by comparing with dose rates on canisters 266 – 270 (which averaged 2,716 R/h) to the dose rates on the two evaluated canisters (which averaged 490 R/h). This calculation shows a decontamination factor of approximately 5.5.

This decontamination process – reducing radionuclide concentrations in the molten glass pool before use of the evacuated canisters – would be expected to remove key radionuclides in approximately the same proportions. Table 4-4, compares scaling factors based on analytical data from the last two batches of HLW slurry feed material with analytical data from samples of glass taken from the evacuated canisters, which represent the material in the vitrification melter at the time it was shutdown. Comparing the glass sample scaling factors to the average of scaling factors from batch 74 and batch 75 shows that Sr-90 was removed in essentially the same proportion as Cs-137 and other key radionuclides were removed in slightly higher proportions than Cs-137. Analysis of input and output for the vitrification melter suggests that scaling factors based on batch 75 are more representative of the radionuclide distribution in the glass in the plugged exit port as can be seen in Table 4-4 below.

[19] The measured concentrations of feed material in the concentrator feed makeup tank are not used for this purpose because of the complex relationship between melter input and melter output. Melter input consisted of HLW slurry – and, for the last two batches, vitrification system decontamination solutions consisting of nitric acid, water, removed waste, and glass formers – that was slowly feed into the melter on a continuous basis. Melter output consisted of molten glass airlifted from the glass pool multiple times to fill the stainless steel canisters. Given the input-output differences, the molten glass pool in the vitrification melter typically contained material from more than one feed material batch.

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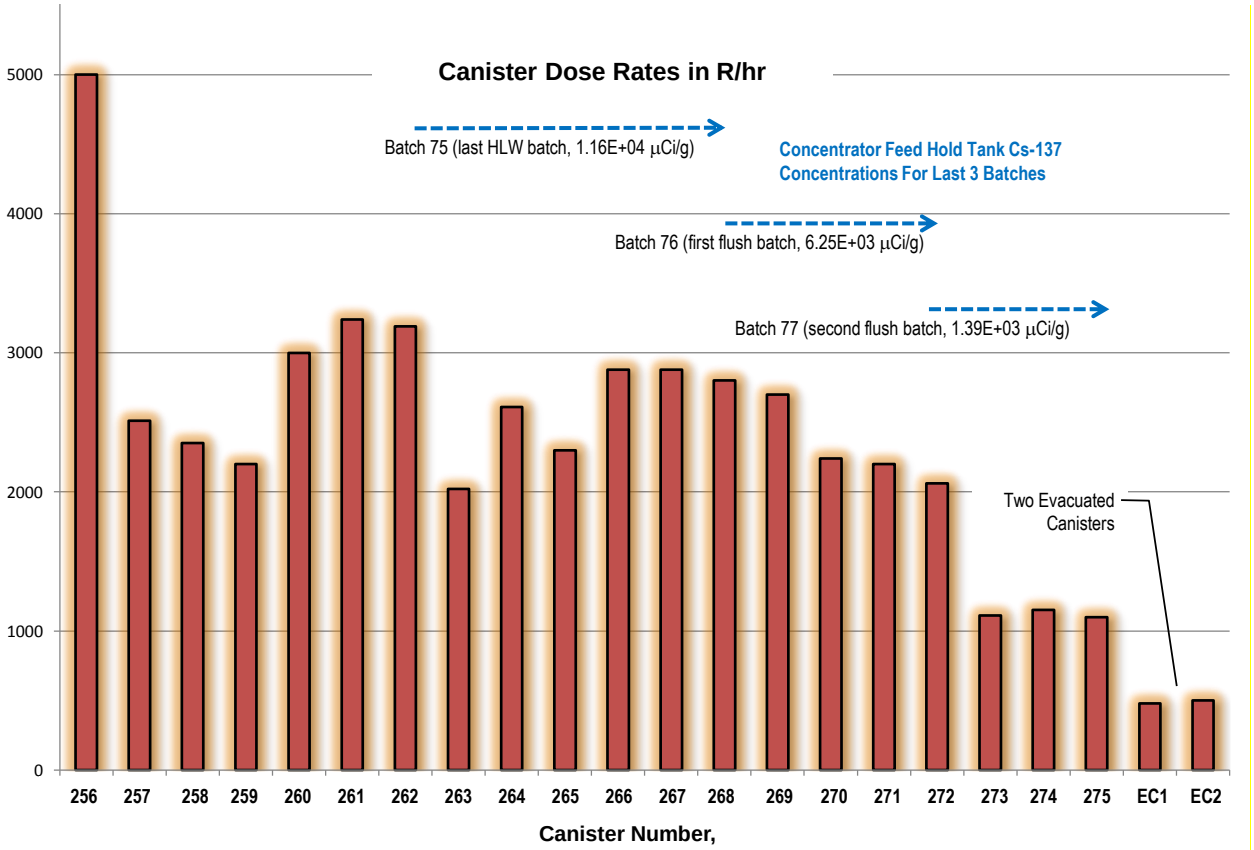

Figure 4-1. Relationship Between Feed Material Cs-137 Concentrations and Canister Dose Rates

Table 4-4. Radionuclide Scaling Factors (Ratios to Cs-137)

Radionuclide	Scaling Factor Basis		
	Glass Sample Data(1)	Batch 74 Data(2)	Batch 75 Data(3)
Sr-90	5.73E-02	4.00E-02	7.47E-02
Np-237	1.41E-06	3.25E-07	6.13E-07
Pu-238	1.57E-04	5.48E-05	1.09E-04
Am-241	6.84E-04	2.19E-04	3.23E-04

NOTES: (1) From WMG 2004b, Exhibit 1.
(2) From concentrator feed makeup tank analytical data shown in the spreadsheet provided to NRC (WVES 2011).
(3) Based on Verification Analytical System Tracking (VAST) sample 01-2498 (WVNSCO 2002a).

The economic practicality of maintaining the vitrification melter operational to process additional vitrification system decontamination solutions is discussed below.

4.2.4 Evacuated Canisters

The WVDP developed the Evacuated Canister System in parallel with the overall Vitrification Facility design to enable removal of the residual dilute molten glass expected to remain in the Vitrification Melter prior to its shutdown and cooling. The Evacuated Canister System consisted of two stainless steel canisters the same size as a standard HLW canister but equipped with a special L-shaped "snorkel" assembly, both of which were designed to be positioned within a steel cage and positioned over the Vitrification Melter. The residual glass was retrieved into the canisters under vacuum after the plug within the snorkel melted from the heat of the Vitrification Melter after its insertion.

Figure 4-1 [figure 4-1 in the draft evaluation] shows the snorkel portion of the evacuated canister in the process of removing molten glass from the Vitrification Melter. The metal has been heated by the molten glass to the point where it produced the orange glow shown in the figure. The lower portion of the snorkel was designed and positioned to reach approximately six inches from the bottom of the Vitrification Melter cavity for the first evacuated canister and approximately two inches from the bottom for the second one.

During deployment, approximately 88 percent of the residual molten glass volume was removed using this system. This technology was technically practical only for removal of key radionuclides from the Vitrification Melter while they were contained in the molten glass. Because the Evacuated Canister System already existed and had been deployed before the time of this analysis, no further economic practicality assessment of this technology was performed.

As a bulk waste removal method that simply removed molten glass from near the bottom of the homogeneous glass pool[20], the Evacuated Canister System removed all key radionuclides from the Vitrification Melter with approximately equal efficiency.

4.3 Economic Practicality Assessment

Economic practicality includes consideration of total lifecycle costs, the cost per curie removed, the relationship between costs and removal of the key radionuclides, and the point in this relationship at which removal costs increase significantly and thus become impractical (DOE Guide 435.1). In this regard, removal of key radionuclides to the "maximum extent . . . economically practical" includes consideration of net social benefit, the remaining service life of equipment, expert judgment, and whether the benefits to health and safety outweigh the disadvantages, that is, whether further radionuclide removal would be useful and sensible in light of the overall benefit to human health and safety.

The evaluation of the economic practicality of additional radionuclide removal from the Vitrification Melter focused on:

(1) Performing additional flushing before shutdown of the Vitrification Melter, considering the effectiveness of the flushing already performed; and

(2) Vitrification Melter dismantlement, in comparison with intact removal and shipment of the intact Vitrification Melter for offsite disposal.

[20] Natural convection currents within the vitrification melter molten glass pool from heat input ensured a homogeneous mixture of the calcined waste and glass formers (Petkus 2002, DesCamp and McMahon 1996).

In each case, the evaluation compared the potential benefits in improved worker and public health and safety (i.e., reduced worker and public risk from radiation exposure) with the expected impacts.

A cost-benefit analysis was performed using methodology consistent with guidance issued by the U.S. Office of Management and Budget (OMB 1992) and by NRC in NUREG/BR-0058, *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission* (NRC 1995) and the companion handbook, NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook*, (NRC 1997). This analysis is documented in a detailed report (Purdue 2004). The following discussion makes use of information from that report.

4.3.1 Evaluation of Additional Flushing

This subsection discusses the results of an evaluation performed to determine the economic practicality of additional vitrification system flushing with regard to decontamination of the Vitrification Melter.

As noted previously, vitrification system flushing included two flushes or decontamination cycles for the Concentrator Feed Makeup Tank and three for the Melter Feed Hold Tank. These flushes were performed in accordance with the flushing plan (WVNSCO 2002a) as part of the system shutdown program developed by the Vitrification Completion Team. They effectively decontaminated the two vessels (WVNSCO 2002c).

The flush solutions were sent to the Vitrification Melter in two batches, batch 76 and batch 77 as shown in Figure 4-1. The flush solutions totaled approximately 220,000 gallons, although more than 95 percent of this amount was evaporated. They effectively decontaminated the melter by reducing radionuclide concentrations in the molten glass pool by a factor of approximately 5.5, as noted previously.

The flushing plan, as noted previously, provided that "flushes may be repeated based on the results obtained." Although the results obtained were satisfactory after a third flush of the Melter Feed Hold Tank, additional flushes of the Concentrator Feed Makeup Tank, the Melter Feed Hold Tank, and/or the entire vitrification system could have been performed before use of the evacuated canisters.

Conceptual Model for an Additional Flush

A simplified model with the following characteristics (Kurasch 2011) was developed to evaluate the economic practicality of processing additional flush solutions in the Vitrification Melter:

- Actual recorded data are used, including radionuclide concentrations from samples of materials in the concentrator feed makeup tank, melter cavity glass levels, canister weights, and canister glass heights.
- Total activity is considered, rather than just Cs-137.
- Progressive cumulative results were produced on a canister-by-canister basis.
- The baseline starting point is canister 267, which contained 31,804 curies and 1,992 kg of glass, with an activity concentration of 15.97 curie/kg of glass. (Canister 267 was filled at the time the last part of batch 75 was being fed to the melter, which indicates that the

radionuclide concentrations in in the melter glass pool were approximately the same as the radionuclide concentrations in this feed material batch.)

- A glass density of 2.4 g/cm^3 is used consistent with the density value used in the vitrification program records.
- One additional batch of flush solutions with glass formers is fed to the melter before use of the evacuated canisters and this feed material batch is nonradioactive for conservatism.
- The actual airlift numbers are used, with fifteen airlifts assumed for filling the additional hypothetical canister, a value consistent with vitrification records.

Figure 4-2 illustrates the conceptual model. Microsoft Excel was used with these inputs to calculate the total activity in the melter glass pool over time.

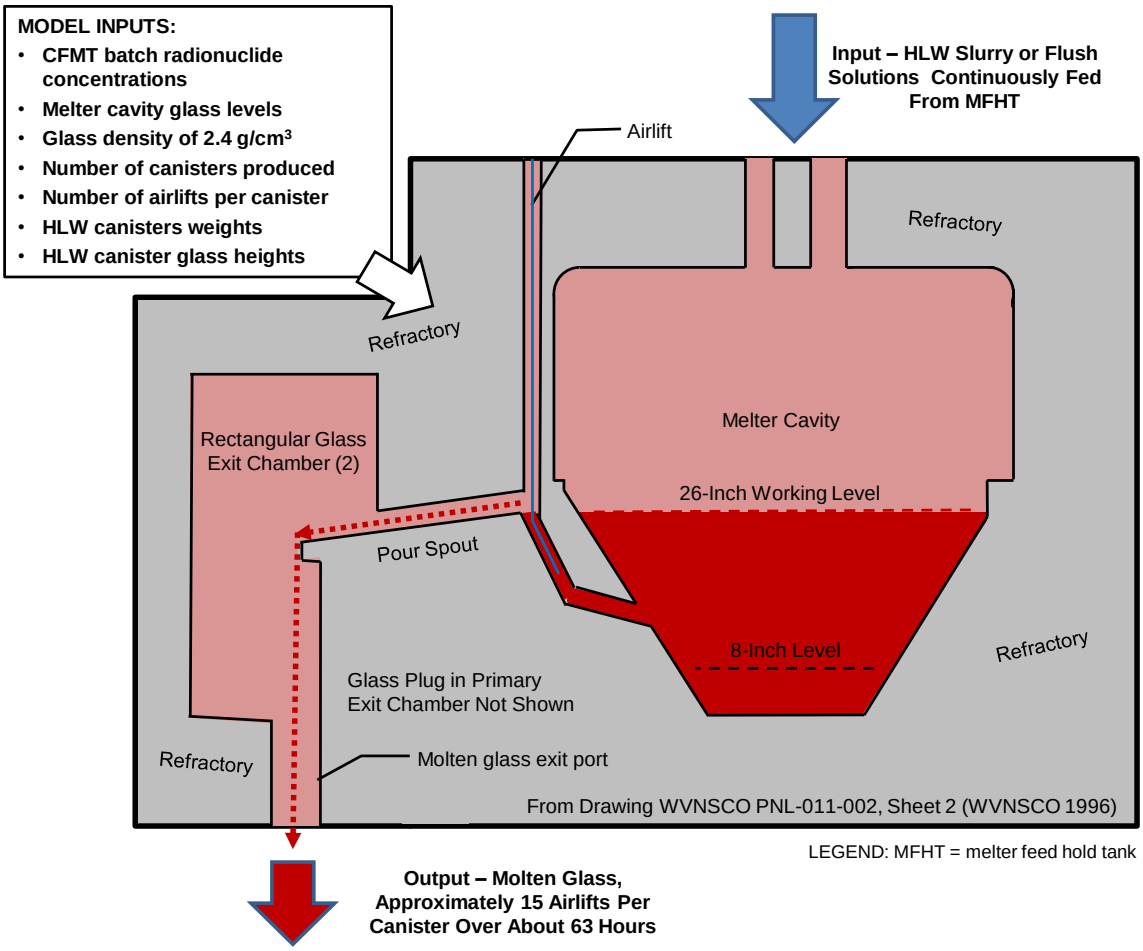

Figure 4-2. Conceptual Model

Results of Implementing the Model

The results predict a radioactivity concentration in the glass pool at the time the actual evacuated canisters were used – that is after batch 77 was fed to the melter – of 2.4 curies/kg. This value is in close agreement with the average of 2.51 curies/kg based on analytical data from glass shard samples taken from the two evacuated canisters. Table 4-5 shows the principle results.

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Table 4-5. Estimated Effectiveness of Processing More Flush Solution Batches(1)

Concentrator Feed Makeup Tank Batch and Last Canister Filled While this Batch Was Being Fed to the Melter	Glass Pool Activity (Ci)
75 (last HLW batch, canister 267)	31,908
76 (first flush solutions batch, canister 272)	16,012
77 (second flush solutions batch, canister 275)	4,303
78 (Hypothetical flush solutions third batch, one hypothetical additional canister)	1,814

NOTE: (1) From Kurasch 2011.

Figure 4-3 displays the model results.

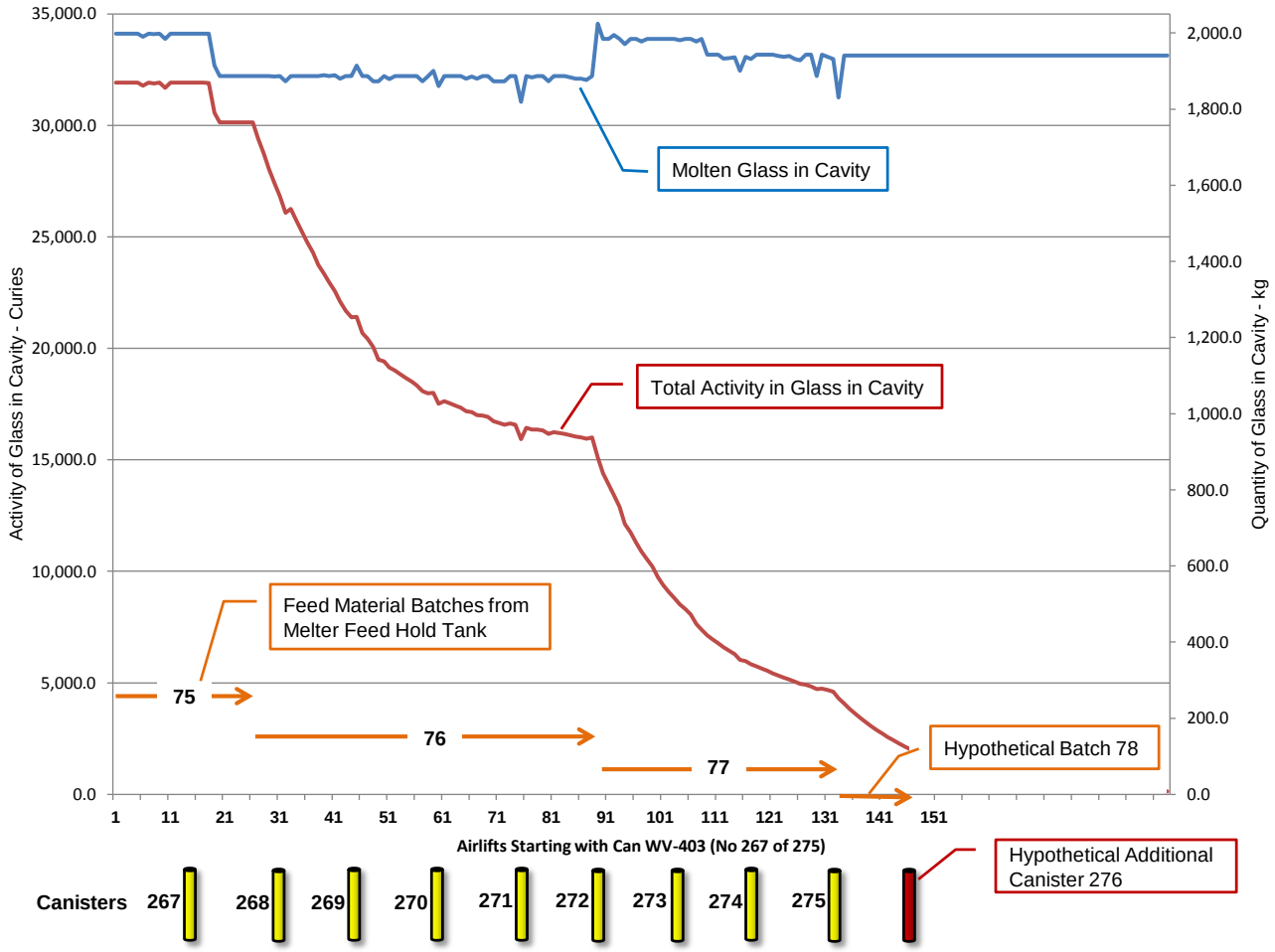

Figure 4-3. Predicted Total Activity in Vitrification Melter Glass Pool

Table 4-5 and Figure 4-3 show that the total activity in the molten glass pool in the Vitrification Melter cavity would have been approximately 1,814 Ci if an additional batch of nonradioactive feed material sufficient to produce a single additional canister has been processed in the melter. If 88 percent of this amount had been extracted by the evacuated canisters, this would have left approximately 218 curies of total activity in the melter cavity. The total activity in the melter then

would have been approximately 500 Ci counting the estimated activity in the plugged discharge port and in fissures in the refractory.

Note that the activity in the 4,303 Ci prediction for the total activity in the molten glass pool before use of the evacuated canisters compares to approximately 5,640 Ci based on the radionuclide concentrations measured in the glass shard samples from the evacuated canisters multiplied by the measured mass of glass in the melter cavity. If the latter estimate were to be used to calibrate the model results, then total activity left in the melter after processing the hypothetical additional flush solutions would be approximately 565 Ci.

Benefits of Processing the Additional Flush Solutions

One benefit would have been realized had the hypothetical additional flush solutions been processed in the Vitrification Melter: The shielded container could have been designed and constructed of lighter weight steel, which would have reduced costs associated with materials, fabrication, and transportation by approximately $100,000.[21]

Consideration was given as to whether other benefits could have resulted from processing of additional flush solutions in the Vitrification Melter. However, other benefits would have been limited for the following:

- The gamma radiation levels on the outside of the waste package in its present condition are low and compliance with radiological control program requirements in handing of the waste package at the WVDP and the LLW disposal facility will ensure the protection of individuals during operations related to disposal as discussed in Section 5.2.4.
- Worker radiation doses would not have been significantly reduced. The external dose rates with a reduced amount of residual radioactivity (and thinner shielding) would have been approximately the same. In any case, the highest dose rate three meters from the waste package, which is located three meters from the bottom, is produced by radioactivity in the plugged primary discharge port; this dose rate would not have been affected by a reduction in the radioactivity inside the melter cavity from processing of additional flush solutions
- The processing of solutions from the flushes actually performed left the melter in a condition suitable for disposal as LLW as discussed in Section 5.3.
- The potential impacts to the general population from disposal of the melter waste package at the LLW disposal facility without further decontamination will be negligible as discussed in Section 5.2.2, so a further reduction in residual radioactivity would not have been beneficial from the standpoint of potential doses to members of the public.
- The potential impacts to an inadvertent intruder from disposal of the melter waste package at the LLW disposal facility without further decontamination will be negligible as discussed in Section 5.2.3, so a further reduction in residual radioactivity would not have been beneficial from the standpoint of an inadvertent human intruder.

[21] This estimate is for a one-time savings. It is a conservative, order-of-magnitude estimate in 2002 dollars for savings in raw material, handling, and transportation costs.

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Costs of Processing the Additional Flush Solutions

A total of approximately $1 million in additional costs (in 2002 dollars) would have been involved. One additional flushing and processing cycle would have taken about two weeks to complete at a cost of approximately $1 million, based on vitrification system operating costs that were running $25 million to $30 million per year[22].

Another factor in considering processing of additional flush solutions was the limited Vitrification Melter service life. Analysis indicated a 35 percent probability of Melter failure occurring within six months of additional operation (Purdue 2004). A catastrophic failure would have, for all practical purposes, stranded radionuclides within the melter since use of the evacuated canister system would not have been feasible with the residual glass in the melter cavity in solid form.

In the interest of conservatism, no attempt was made to quantify other costs associated with processing of additional flush solutions, such as the monetary value of additional worker radiation dose that would have been necessary to continue vitrification melter operations, which was estimated to be approximately seven person-rem per month (Perdue 2004).

Conclusions

The foregoing discussions show that processing of additional flush solutions in the Vitrification Melter to produce a single additional canister would have produced negligible, if any, benefits in terms of improved worker or public health and safety and would have resulted in approximately $1 million in additional monetary costs.

Additional References

Kurasch 2011, *Calculation of the Results of Processing Additional Flush Solutions in the WVDP Vitrification Melter Based on Actual Vitrification Records Data*. Kurasch, D., West Valley Environmental Services, West Valley New York, June 2011.

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[22] In addition, there would be indirect life-cycle costs. Interim onsite storage of one additional canister would have cost around $35,000 per year based on actual and predicted annual costs of interim storage, including surveillance and maintenance.

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WVNSCO 2002c, *Letter from J. Paul of WVNSCO to A. Williams of DOE-West Valley on completion of Fiscal Year 2002 contract milestone on Melter Feed Hold Tank Ffushing.* WV 48, WD:2002:0400, West Valley Nuclear Services Company, West Valley, New York, July 30, 2002.

WVNSCO 2002d, *Letter from J. Paul of WVNSCO to A. Williams of DOE-West Valley on completion of Fiscal Year 2002 contract milestone on completion of HLW vitrification operations.* WV 48, WD:2002:0400, West Valley Nuclear Services Company, West Valley, New York, July 30, 2002.